Advanced Fuel Cycle Initiative	**DOCUMENT RELEASE AUTHORIZATION**	

Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

ANL AFCI Document Number:	ANL-AFCI-100	Release Date:	9/23/2003

Title:	Systematic Assessment of LWR Recycle Strategies
Author(s):	M. Salvatores, G. Youinou, R. N. Hill, T. Taiwo, T. K. Kim

Approval for Release

Approved by:			**Date**
Principal Author:	Massimo Salvatores		9/19/03
ANL Work Package Manager:	Robert N. Hill	Robert N. Hill	9/22/03
ANL Program Manager:	Phillip J. Finck	PF Finck	9/22/03

1. Introduction

The long-lived waste transmutation with the objective to lower the burden on a repository as far as:

- masses/volumes
- heat load
- doses/radiotoxicity

has been studied both in fast neutron systems and in thermal neutron systems.

Findings related to transmutation in LWRs are summarized in the present report. The results are mainly based on past and current studies, performed in Europe, USA, Japan and in the frame of international organisations. The indications given here are focused essentially on standard LWRs with more or less standard fuel. Information and studies available allow a fairly systematic approach (sections 2 and 3).

More innovative options, e.g. inert matrix fuels, are also treated, both for Pu burning and for the heterogeneous recycling of minor actinides (sections 4 and 5), but more in the form of a survey of the somewhat limited material available.

Finally, some indications will be given in section 6 of the use of other thermal spectrum reactors for Pu (and sometimes minor actinides) management.

A comparative study based on a consistent methodology is described in Section 7, which will bring more quantitative effects regarding the more innovative concepts, and which will be the object of a future study.

2. Transmutation in LWRs

2.1 INTRODUCTION

In general, it has been shown that to reach the objective of reduction of heat load, doses/radiotoxicity in a repository the Pu and minor actinide (MA) inventory of spent fuel must be reduced, one can use two approaches:

- Use of multirecycling of the fuel loaded with Pu and MA. In this case the “key” parameter which defines the performance achievable with transmutation is the **“separation efficiency”** for Pu and MA (e.g. 0.1 % losses to the wastes).
- Use of long irradiation time for Pu and/or MA loaded fuels/targets in a once-through-then-out (OTTO) approach. In this case the “key” parameter is the **“cumulative fission rate”** (e.g. 90 % or more of fissions in the fuel/target).

Two fuel forms can be considered for the approaches. These are:

- “Homogeneous” fuels, where Pu and MA are closely mixed/dispersed (homogeneous recycling).
- Targets (or dedicated, i.e. U-free, inert matrix fuels) of MA, generally dispersed in an inert matrix (heterogeneous recycling).

For Am and Cm, also two strategies can be envisaged:

- Am only in the fuel, with Cm sent to the repository or to intermediate storage.

- Am and Cm together in the fuel.

As far as Np, two strategies can also be used:

- Np mixed with Pu (e.g. not been separated at fuel reprocessing).
- Np added to Am or to Am+Cm, according to what is indicated above.

2.2 THE PHYSICS OF THE "MULTIRECYCLING" APPROACH

Because plutonium is the source material for MA generation (via either decay or capture reactions), the first requirement is to adapt any MA strategy to the Pu management approach. This means that a strategic choice has to first be made on the Pu build-up in LWRs of the present generation, i.e. Pu build-up decrease, e.g. up to a constant value to be defined, or a further decrease with respect to an "equilibrium" value.

If the recycling of Pu in LWRs is chosen, there is a basic difficulty in Pu "multirecycling", related to the degradation of the Pu vector.

The recycling of plutonium as PuO_2/UO_2 mixed oxide (MOX) fuel is already established in pressurized water reactors (PWR) in several countries on a commercial scale. The discharge burn-up of MOX fuel, and indeed its overall performance, is essentially the same as that of UO_2 fuel. Thus the MOX fuel currently being irradiated in PWRs is typically intended to be discharged at burn-ups of 40 to 45 MWd/t. Extension to even higher burn-ups (e.g. 60 MWd/t) is not expected in very near future. The initial plutonium content needed to achieve such burn-ups varies depending on the precise source of the plutonium, but is typically in the range 7 to 8 w/o total plutonium, expressed as an average over the whole assembly. The experience of MOX utilization in PWRs has been positive and there are no outstanding operational or safety issues to be resolved. However multirecycling beyond the first recycle presents new challenges.

For thermal reactors the even isotopes of plutonium (238, 240 and 242) do not contribute significantly to fission and the ratio ($^{239}Pu + {}^{241}Pu$)/(total plutonium) denotes the fissile fraction of the plutonium and is a measure of plutonium quality for thermal reactor MOX. The primary problem is that plutonium quality decreases as the discharge burn-up increases and decreases yet further following recycle of the plutonium recovered from MOX. Combined with the need to increase plutonium content to reach higher burn-ups, it will be necessary to significantly increase the total plutonium content of the MOX fuel during multiple recyclings.

Compared with conventional UO_2 fuel, MOX fuel is already significantly different from a neutronic point of view, there being a much smaller thermal flux for a given rating. This is due to the combined effects of the higher fission and absorption cross-sections of ^{239}Pu and ^{241}Pu compared with ^{235}U, exacerbated by the significant absorption of the ^{240}Pu. The difference in spectrum affects the core performance because the control, reactivity coefficient and transient behaviors are all altered. Increasing the total plutonium content beyond present levels exaggerates all these effects further. Ultimately, the deterioration in parameters such as control rod reactivity worth, boron reactivity worth and moderator void and temperature coefficients may become a barrier to further utilization of MOX in PWRs, at least in conventional lattices.

The results of a series of extensive international benchmark (Ref. 1) indicated that the moderator void coefficient is the first barrier to show-up during successive recyclings.

The void coefficient is very important for safety and should be negative, or at least non-positive to ensure negative feedback. The void coefficient tends to become less negative the

higher the total plutonium content and in the conventional PWR lattice changes sign from negative to positive at a total plutonium concentration of between 10 and 12 w/o, which is reached at the second recycling, whatever the quality of the initial Pu loading.

In fact, the need to increase at each recycle the Pu content in the fuel, induces a hardening of the neutron spectrum, up to the point when the coolant void coefficient becomes positive due to the reduced absorption in the eV range resonances of Pu-240 and Pu-242.

The benchmark comparisons showed reasonably satisfactory level of agreement between the various solutions submitted and a wide consensus was reached on that barrier for further recyclings.

Moreover, since the multirecycling of Pu induces a relevant increase of the MA production (with respect to the OTTO strategy with UOX), one has also to minimize the ratio :

MA production/Pu consumption

From a physics point of view, this ratio is closely related to cross-section ratio $\alpha = \bar{\sigma}_c / \bar{\sigma}_f$ of the TRUs and its variation with the hardness of the spectrum.

In order to vary the spectrum hardness, one can modify the V_m/V_f moderator-to-fuel ratio.
To stay in a realistic range, the V_m/V_f ratio can be varied between ~ 1 and 4 (standard PWR value: ~ 2). In what follows, we will summarize the main results obtained varying the V_m/V_f parameter.

2.2.1 The effect of the moderator-to-fuel ratio

It is a well known fact that the spectrum dependence of cross-sections gives rise to large differences in one-group cross-sections between e.g. PWRs and fast neutron reactors (FNR), but also among PWRs with different ratios of moderator-to-fuel volume V_m/V_f. Higher V_m/V_f ratios imply more thermalized spectra. In general, the capture/fission ratio is higher at thermal energies (increases with moderation) particularly for the fertile species; this behaviour is illustrated in Table I:

TABLE I. Ratio of σ_c/σ_f for different types of spectra

	PWR			
	V_m/V_f = 3	**= 2**	**= 1.1**	**FNR**
U235	0.236	0.317	0.403	0.312
U238	7.39	6.42	5.62	8.65
Np237	51.9	34.0	21.2	5.74
Pu238	10.5	4.77	2.30	0.552
Pu239	0.546	0.561	0.564	0.337
Pu240	77.0	40.0	16.8	1.89
Pu241	0.330	0.318	0.292	0.207
Pu242	33.2	23.2	12.9	3.23
Am241	74.3	48.7	27.9	8.55
Am242	0.232	0.220	0.194	0.211
Am243	89.3	69.2	44.1	9.82
Cm242	4.03	4.21	3.97	1.067
Cm243	0.183	0.175	0.176	0.074
Cm244	15.6	14.1	11.1	1.53
Cm245	0.147	0.147	0.152	0.127

Moreover, the irradiation length (burn-up) of the UOX fuel has to be taken into account, since it has influence on:

- the Pu isotopic vector,
- the amount of Pu which builds up.

The higher the burn-up, the worse the Pu "quality" becomes (i.e. less fissile isotopes, Pu-239, Pu-241, and more Pu-242) as shown in Table II. The first set of results in Table III (kg per initial heavy metal mass) shows that the Pu and MA content of spent PWR fuel increases with burnup. However, the higher burnup allows a reduction in the heavy metal feed rate. The second set of results in Table III (kg per energy) shows that the plutonium generation for a given energy production actually decreases with burnup because the plutonium inventory is slowly saturating. Conversely, the Am-243 and Cm-244 continue to build-up even with the energy normalization as their generation rates (e.g., Pu-242 capture) are still increasing rapidly due to accumulation of the source isotopes.

TABLE II. Plutonium isotopic composition from spent UOX fuel (after 5 years decay)

Burn-up (GWd/t) of the UOX fuel	Pu 238	Pu 239	Pu 240	Pu 241	Pu 242
33	1.8	57.9	22.5	11.1	5.6
42	2.7	54.5	22.8	11.7	7.0
55	4.0	50.4	23.0	12.3	9.1
65	4.8	47.5	23.8	12.1	10.5

TABLE III. Pu and Minor Actinide Generation from spent UOX fuel (after 5 years decay)

BURN-UP (GWd/t)	Initial Enrichment	TOTAL Pu	Np 237	Am 241	Am 243	Cm 244	Cm 245
		kg/MTIHM					
42	3.70 %	11.7	0.62	0.20	0.17	0.05	0.00
55	4.50 %	12.6	0.76	0.21	0.25	0.10	0.02
65	4.95 %	13.3	1.01	0.25	0.38	0.18	0.02
		kg/TWeh					
42	3.70 %	34	1.81	0.58	0.49	0.14	0.01
55	4.50 %	28	1.70	0.47	0.56	0.22	0.02
65	4.95 %	25	1.90	0.48	0.71	0.34	0.03

When MOX fuel is used in a PWR, Am-241, Am-243 and Cm build-up is exacerbated with respect to UOX fuel irradiation. With multiple recycling, the build-up of these minor actinides grows significantly as shown in Table IV and the Pu "quality" becomes progressively worse (see Table V). This spoiling of the plutonium vector can be slowed by dilution of the MOX recycle plutonium with UOX plutonium. As shown in Table V, this approach also mitigates the required increases in the MOX enrichment which have safety impacts as shown in Section 2.2.2.

TABLE IV. MINOR ACTINIDE PRODUCTION (kg/TWeh) IN PWRs WITH UOX AND MOX FUEL (65 GWd/T burnup, after 5 years decay)

Type of fuel	Np 237	Am 241	Am 243	Cm 244	Cm 245
MOX	0.5	8.6	5.7	3.3	0.7
UOX	1.9	0.48	0.71	0.34	0.03

TABLE V. Pu recycling in a PWR-MOX with standard moderator-to-fuel ratio (= 2), (burn-up 50 GWd/t, 5 years cooling)

	Recycling with dilution			Self-recycling		
Number of recyclings	**1**	**2**	**3**	**1**	**2**	**3**
Fissile Pu content (%)	6.0	6.6	7.0	6.0	7.6	8.6
Pu vector (%) at beginning of cycle:						
Pu-238	4.0	4.5	5.0	4.0	5.5	6.7
Pu-239	50.4	45.4	42.2	50.4	37.5	31.7
Pu-240	23.0	25.8	26.9	23.0	28.3	29.7
Pu-241	12.3	12.6	12.7	12.3	13.4	12.4
Pu-242	9.1	10.4	11.9	9.1	13.9	18.0
Total Pu	9.5	11.3	12.7	9.5	14.9	19.5

By varying the moderator-to-fuel ratio, the neutron energy spectrum of the PWR can be significantly altered; this induces changes in the capture-to-fission ratios as shown in Table I. As shown in Table VI, these spectral variations lead to large differences in the plutonium enrichment requirements with the reduced moderation concept requiring more than twice the enrichment of the highly moderated design. The high plutonium content leads to corresponding increases in the minor actinide production rates as shown in Table VII.

TABLE VI. Pu recycling in PWR-MOX with different moderator-to-fuel ratios, (burn-up 50 GWd/t, 3-years cooling)

V_m/V_f	= 3			= 2			= 1.1		
Number of recyclings	**1**	**2**	**3**	**1**	**2**	**3**	**1**	**2**	**3**
Total Pu (%)	6.7	8.7	10	9.6	11.3	12.7	14	14.7	15.3
Fissile Pu (%)	4.2	4.6	5.0	6.0	6.6	7.0	8.8	8.9	9.0
Fissile Pu/Total Pu (%)	62.7	53.6	50	62.7	58	54.9	62.7	60.6	59.0
Pu vector (%) at beginning of cycle :									
Pu-238	4.0	4.5	4.9	4.0	4.5	5.0	4.0	4.2	4.3
Pu-239	50.4	41.5	37.8	50.4	45.4	42.2	50.4	49.1	47.8
Pu-240	23	27.3	28.0	23	25.8	26.9	23	24.9	26.1
Pu-241	12.3	12.2	12.2	12.3	12.6	12.7	12.3	11.5	11.2
Pu-242	9.1	13.3	15.9	9.1	10.4	11.9	9.1	9.1	9.5
Am-241	1.2	1.2	1.2	1.2	1.3	1.3	1.2	1.2	1.1

TABLE VII. Minor Actinide production (kg/TWhe)
Values at the end of cycle and after 3 years cooling, burn-up of 50 GWd/t)

	PWR-MOX 100 % Moderator-to-fuel ratio: 3			PWR-MOX 100 % Moderator-to-fuel ratio: 2			PWR-MOX 100 % Moderator-to-fuel ratio: 1.1		
Number of recyclings	**1**	**2**	**3**	**1**	**2**	**3**	**1**	**2**	**3**
Np-237	0.29	0.3	0.3	0.36	0.36	0.35	0.43	0.43	0.43
Am-241	3.0	4.7	5.8	6.7	8.4	9.7	10.7	11.1	11.5
Am-243	4.7	6.3	7.3	5.6	6.5	7.3	6.5	6.7	6.9
Cm	2.9	3.5	3.9	3.7	4.0	4.2	3.9	3.9	4.0
Total MA	10.9	14.8	17.3	16.4	19.4	21.7	21.8	22.3	23.0

The net effect on Pu and minor actinide inventories are summarized in Table VIII. Because Am-241 is produced by Pu-241 decay, its production rate will be proportional to the plutonium inventory. However, the production of higher actinides (through the Pu-242 to Am-243 gateway) will be impacted by the spectral differences noted in Table I. Thus, the production rate of minor actinides for a given plutonium inventory is reduced for the low moderation concepts, as shown in the final column of Table VIII.

TABLE VIII. Masses at the end of each recycle and after 3-years cooling (burn-up : 50 GWd/t)

	Pu (kg/TWhe)	M.A. (kg/TWhe)	M.A./Pu-initial (kg/MTIPu)
PWR-UOX	29.3	3.78	
PWR-MOX (V_m/V_f = 3) Cycle 1	-68	10.9	66.4
Cycle 2	-76	14.8	69.4
Cycle 3	-80	17.3	70.6
PWR-MOX (V_m/V_f = 2) Cycle 1	-64	16.4	69.7
Cycle 2	-69	19.3	70.0
Cycle 3	-73	21.7	69.7
PWR-MOX (V_m/V_f = 1.1) Cycle 1	-54	21.8	63.5
Cycle 2	-56	22.3	61.9
Cycle 3	-58	23.0	61.3

The following remarks can be made (Ref. 1)

- *Plutonium*

  The plutonium mass balance is negative for both the Standard (STD) and High Moderation (HM) cases, due to the destruction of the odd fissile isotopes. For both the STD and HM cases, the plutonium destruction rate increases with recycle generation (being some 40 % higher in the fifth generation than in the first generation). In terms of kg/TWh(e), the destruction rate is only marginally higher in the HM case. However, this does not imply that the STD and HM cases are equivalent in terms of plutonium destruction. The point to remember is that the fuel mass is considerably smaller for the HM PWR and, at least in the early recycle generations, the concentration of plutonium is much smaller. Therefore, with reference to the initial plutonium content, the HM PWR is more effective at destroying plutonium.

- *Americium and Curium*

  The mass balances for Am and Cm are positive, meaning that they accumulate with burn-up and the build-up rate is higher in the later recycle generations. This result applies both to the STD and HM cases and is a result of the higher initial plutonium concentrations needed in later recycle generations; the higher initial plutonium mass leads to accelerated production of the trans-plutonium nuclides. In terms of Kg/TWhe, the production rate is lower for the HM PWR than for the STD PWR in the early recycle generations, though the difference becomes marginal in the later generations.

Finally, the apparent conflict between the Table VII results (more minor actinides produced at reduced moderation ratio) and the Table VIII results (fewer minor actinides per plutonium

inventory at reduced moderation ratio) can be explained by considering the fuel cycle options. If the goal is to contain a given amount of plutonium in the LWR fuel cycle, nearly twice as many high moderation MOX reactors will be required; and this will overcome the MA production (per TWeh) advantage shown in Table VII. On the other hand, if the fuel cycle goal is to produce the maximum energy in PWRs while limiting the MA production, the high moderation concept allows reduced enrichment extending the plutonium energy resources.

2.2.2 The physics issues of the moderator void coefficient

The high fissile content necessary for multiple recyclings can have consequences on reactivity coefficients. For example, the limit for the Pu content to avoid a positive moderator void coefficient, is approximately 10-12 % (dependent on the "quality" of the Pu vector).

This is due to the shape of the η of Pu-239 and to the resonance structure of Pu-240 and Pu-242 capture cross-sections (see Figs. 1, 2 and 3) and their effect on the neutron spectrum (see fig. 4).

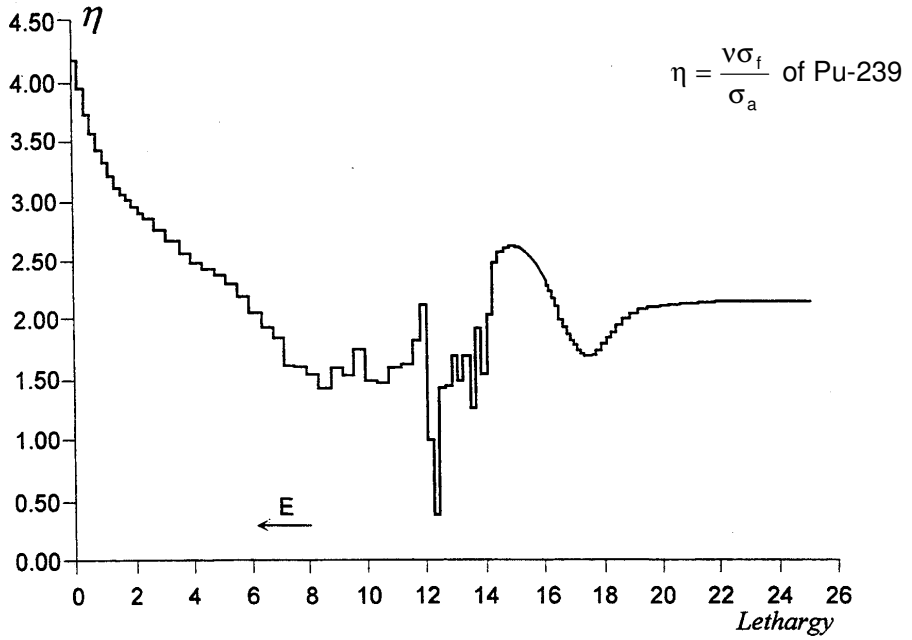


FIGURE 1. Energy variation of eta (neutron production/absorption) for Pu-239

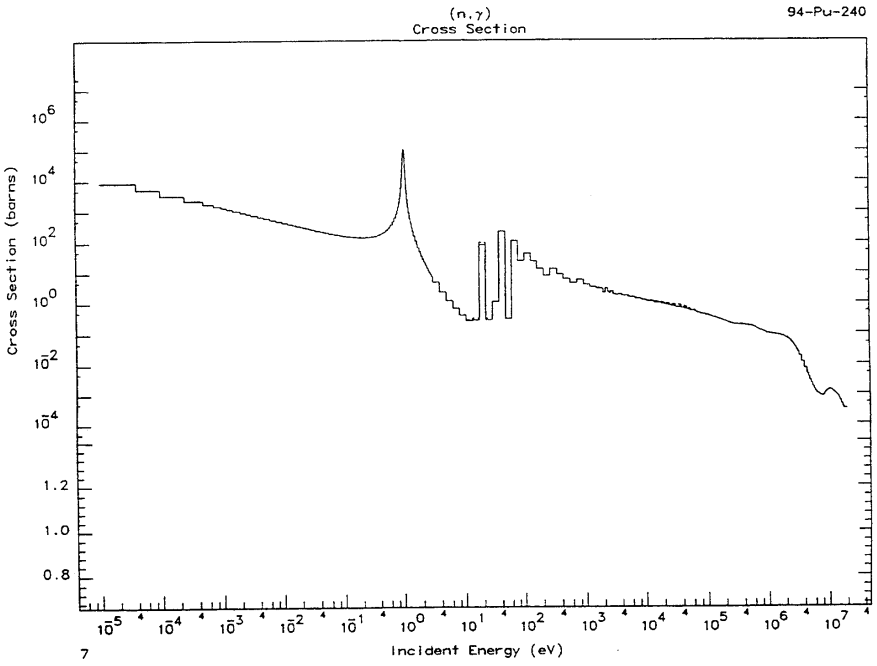


FIGURE 2. Energy variation of Pu-240 capture cross section

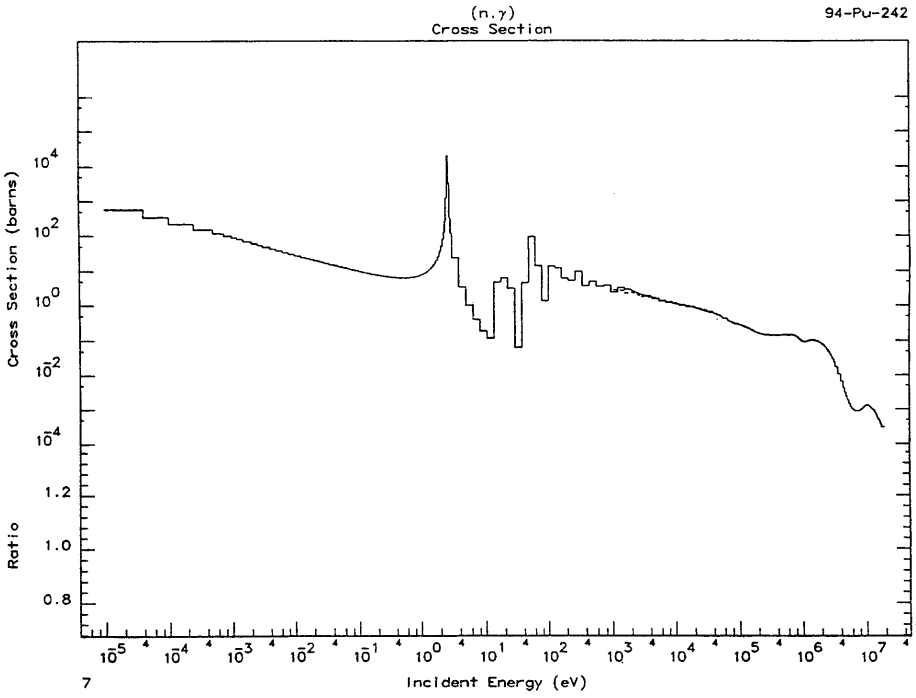


FIGURE 3. Energy variation of Pu-242 capture cross section

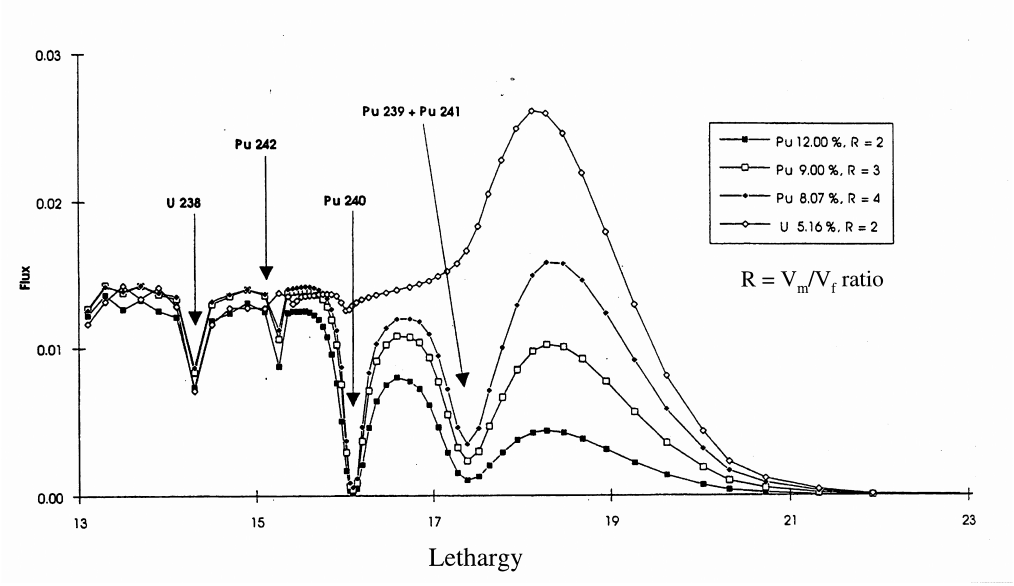


FIGURE 4. Neutron energy spectrum for typical MOX and UOX PWRs

To illustrate the effect of the Pu vector on the void reactivity coefficient, Fig. 5 gives the contribution of different isotopes in a MOX fuel to the reactivity as a function of the void fraction.

It different Pu vectors are used, see e.g. Table IX, the void reactivity varies as a function of the fissile Pu content as shown in Figure 6. It is useful to recall that the first MOX recycle has a vector similar to the Pu no.3 of Table IX.

Table X gives the limits in terms of fissile Pu content to avoid a positive void reactivity coefficient for the different Pu vectors indicated in Table IX.

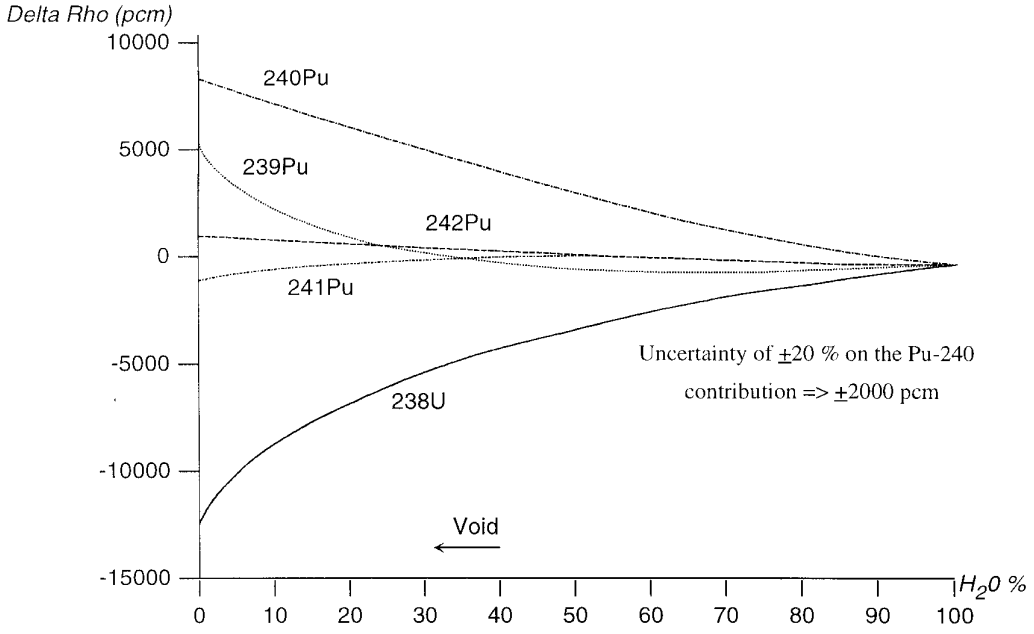


FIGURE 5. Contribution of different isotopes to void reactivity coefficient

TABLE IX. Pu types/Vectors

	Pu n°1	Pu n°2	Pu n°3	Pu n°4	Pu n°5	Pu n°6
^{238}Pu	0.11	1.17	1.85	2.55	2.74	5.63
^{239}Pu	79.93	67.85	58.05	54.26	42.51	33.94
^{240}Pu	17.25	18.63	22.55	23.16	29.19	29.10
^{241}Pu	1.45	9.11	10.75	11.71	14.30	13.71
^{242}Pu	0.50	2.69	5.60	7.14	9.82	16.23
^{241}Am	0.57	0.55	1.20	1.18	1.44	1.39
Pu_{Fis}	81.38	76.96	68.80	65.97	56.81	47.65

In the case of a highly moderated lattice, there is some improvement, which corresponds potentially to one further recycling without reaching a positive moderator void reactivity effect. However, as we have seen previously, the quicker degradation of the Pu quality with multirecycling when $V_m/V_f > 2$, makes this advantage only a marginal one.

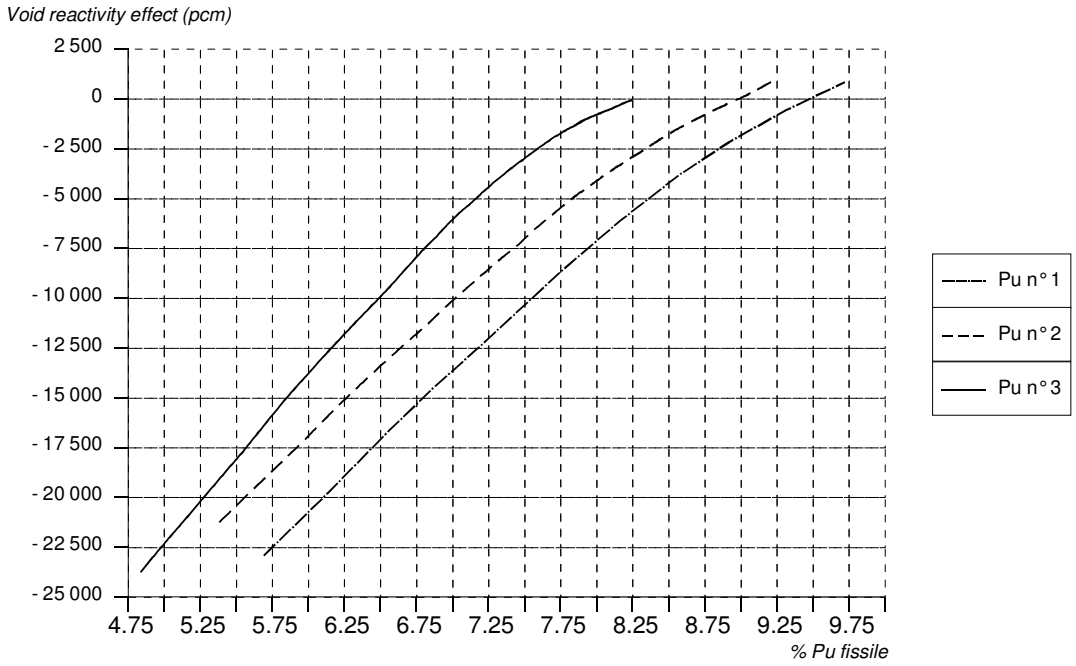


FIGURE 6. Total void reactivity effect for Pu n°1, 2 and 3 as a function of Pu fissile content (%)

TABLE X. Plutonium limit to have a negative void coefficient (in %)

Pu type	Max Pu_{fiss}
n°1	9.7
n°2	9.2
n°3	8.4
n°4	7.1
n°5	6.2
n°6	5.3

In summary, whatever the optimisation, it seems that there are limitations to multiple recyclings of MOX fuel in PWRs, if measures are not taken, as we will see in the next paragraph. In fact, only fast reactors offer the most flexible and straight forward way-out:

- Better σ_f/σ_c ratios for most actinides.
- Better overall neutron economy, which means flexibility to burn (Pu and minor actinides) or to breed.
- Lower production of MA during multirecycling (see table XI).

However, improved concepts of PWRs assemblies can extend the number of recycle achievable, allowing a larger fraction of the Pu and MA inventory to be consumed in such systems.

TABLE XI. MINOR ACTINIDES PRODUCTION IN PWRs AND FBRs (kg /TWeh, single recycling of Pu in PWRs)

			Np 237	Am 241	Am 243	Cm 244	Cm 245
Under moderated PWR	: MOX	42 GWd/t	0.72	13.7	5.64	2.5	0.46
Standard PWR	: MOX	42 GWd/t	0.61	8.2	4.26	2.0	0.29
Highly Moderated PWR	: MOX	42 GWd/t	0.48	4.7	4.15	1.7	0.15
Fast Neutron Reactor	:	125 GWd/t	0.32	3.6	1.41	0.31	0.026

3. Possible implementation of Pu and MA recycling in PWRs

3.1 POSSIBLE STRATEGIES/CONCEPTS

Recent studies indicate that one can envision three different recycling strategies in PWRs, which avoid the known difficulties to multi-recycle Pu in a PWR [see previous paragraph and Ref. 1], and which could allow in principle the multi-recycling of MA and to reach high burn-ups :

a. Pu and MA are mixed with enriched Uranium in an oxide form. This fuel is loaded in standard PWR assemblies over all the core and in all the PWRs of the NPP (MIX option, Ref. 2).

b. Pu and MA are mixed with depleted U in an oxide form. These fuel pins are loaded at the periphery of a standard UOX-PWR assembly (CORAIL option, Ref. 3). Most work on the CORAIL concept has been also performed at ANL, and detailed results, in excellent agreement with the CEA results have been obtained [4].

c. Pu and MA are mixed in a fertile-free matrix (e.g., (Pu/MA Ce)O_2). Pins of this fertile-free fuel are put in standard UOX-PWR assembly, the ratio of these (Pu + MA) pins to standard UOX pins being approximately 20% (APA: Advanced Plutonium Assembly). This concept was initially developed in order to burn Pu (Ref. 5).

More recently, an optimized version of the MIX concept has been proposed (called MOX-UE), in which the maximum Pu content is imposed and U-235 is added starting at the second recycling, growing at successive steps of recycling. This concept, together with a limited increase of the moderator-to-fuel ratio is called MOX-UE RMA (Ref. 6).

More specifically, as summarized by G. Youinou (Ref. 7) the way to circumvent this difficulty is to compensate the degradation of the plutonium quality by adding U-235 in order to maintain a certain reactivity level. That allows the plutonium content to remain at a value ensuring a negative void coefficient even in the case of degraded plutonium, and at the same time, by the addition of U-235, to reach burn-ups of around 60 GWd/t. Hence, different

options mixing slightly enriched uranium together with Pu fuel in standard LWRs were investigated.

The CORAIL option (Figure 7a) utilizes only existing UOX and MOX fuel fabrication technology. The assembly is made up of about two thirds of UOX rods on the inside, and of about one third of MOX rods at the periphery, since this design seemed to be a fair compromise between Pu mass balances and core control parameters. However, because the two fuels have very different neutronic characteristics special attention has to be given to the power distribution at the interfaces. In practice, this means that the U-235 enrichment in the UOX rods, the Pu content in the MOX rods, and its isotopic composition can not be defined independently. The assembly has an approximately zero Pu mass balance, i.e. the Pu burned in the MOX rods is about compensated by the Pu produced in the UOX rods.

Like CORAIL, the MOX-UE option (Figure 7b) makes use of only UOX and MOX technology. The idea is to enrich the uranium matrix of the MOX fuel (hence the name MOX-UE), when the Pu is degraded. Since the assembly is made up of the same kind of rods, there are no specific power distribution issues, which makes the core management easier. With Pu content fixed at around 12%, the Pu mass balances are around -70 kg/TWhe, which corresponds to about 30% of the initial quantity. On the other hand, the control devices have to be modified to increase their efficiency (use of enriched boron in the moderator and in the control rods).

In the APA (Advanced Plutonium Assembly) option (Figure 7c), 144 standard UOX fuel rods are substituted by 36 large annular rods containing Pu in an inert matrix. This design provides a moderation ratio, averaged over the whole assembly, larger than in a standard 17×17 assembly (respectively 3.5 and 2). Furthermore, the presence of the inert matrix in place of U-238 reduces the formation of plutonium by capture on this isotope. These two characteristics make APA very attractive in terms of plutonium incineration : at the end of the irradiation, about half of the plutonium initially loaded has been destroyed either by fission or capture (~ -80 kg/TWhe). This concept (together with other Inert Matrix Fuel concepts) will be further discussed in Section 5.

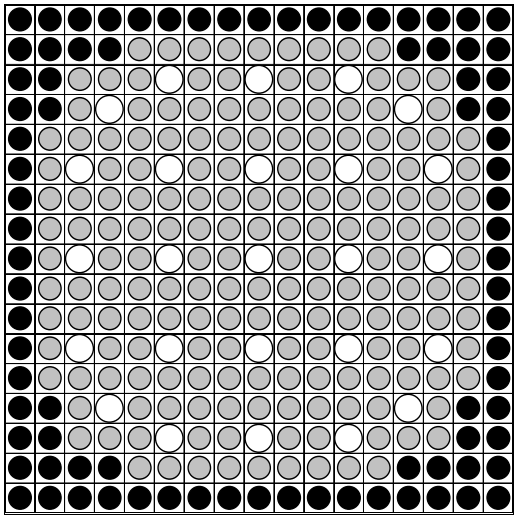
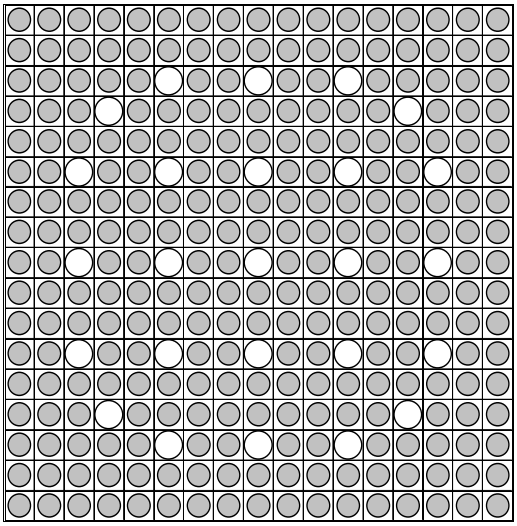


Figure 7a : CORAIL assembly. 84 MOX rods are located at the periphery and 180 UOX rods in the inside.

Figure 7b : MOX-UE assembly. All the rods are similar (PuO2 on an enriched U matrix)



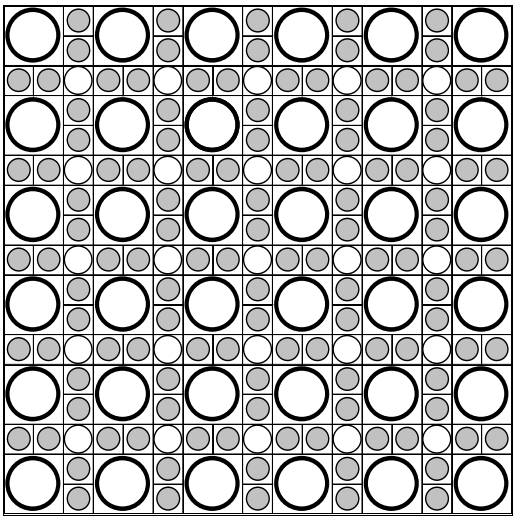


Figure 7c : APA assembly. The Pu is located in 36 large annular rods. There are 120 UOX rods.

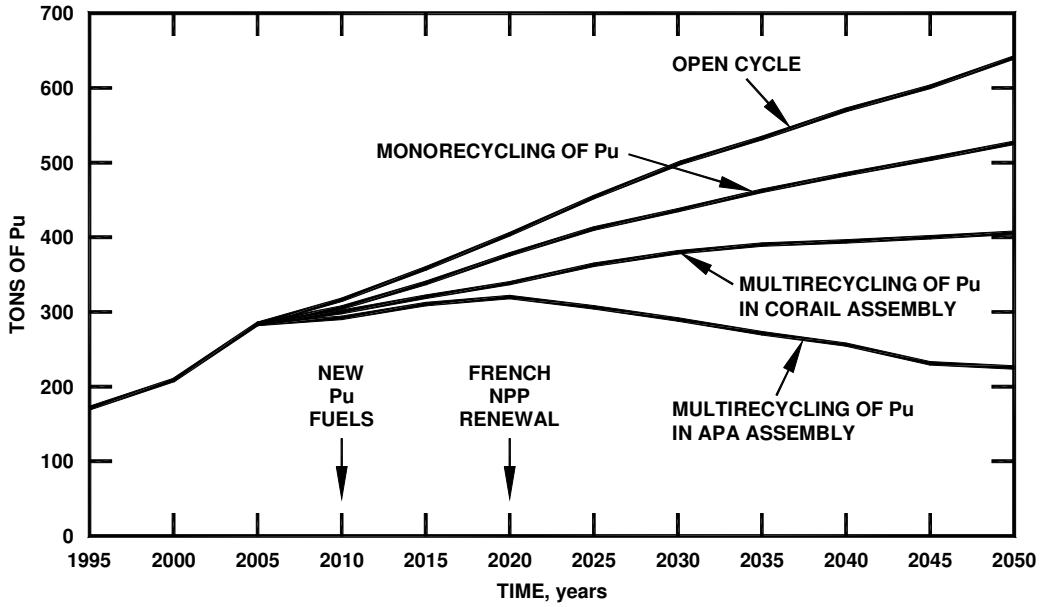
Finally, there is also the option of reducing the pitch in the assembly, reducing the moderator-to-fuel ratio (down to ~ 1.3) and multirecycle Pu (and possibly MA) in it. This option can be foreseen, if one wants to slow-down the consumption of Pu (see Table VIII), preparing the way to the successive introduction of fast reactors, which will need sizable amount of Pu to be made available.

In practice, one could characterize the multirecycling options according to the objective:

- To stabilize the Pu and MA inventories (MOX-UE, MIX, CORAIL).
- To reduce the Pu and MA inventories (APA, but also other Uranium-free inert matrix fuel concepts, as proposed e.g. in Ref. 8 and which will be discussed in detail in a later section).
- To let (moderately) increase the Pu inventory (tight-pitch lattices).

The effect of using the CORAIL and APA assemblies on the Pu stockpile stabilization or reduction is given for the French case, in Figure 8.

FIGURE 8. Pu Stockpile Control with Pu Management in PWRs (the French Case)



Assembly and full core calculations have been performed to show the feasibility of the different concepts [2,3,4,5,6]. The "price" paid, is the additional ^{235}U enrichment needed to compensate the lack of neutrons. It has been shown that all relevant safety and operation criteria can be met (boron effectiveness, reactivity coefficients, power peaking, etc.). Fuel cycle calculations have also been performed in order to establish equilibrium (or "near" equilibrium) compositions, to verify the feasibility of the multiple recycling.

When losses at reprocessing are defined to be 0.1% both for Pu and for MA, radiotoxicity reduction factors, with respect to the open once-through cycle, have been obtained together with annual NPP mass balances, fabrication needs and total inventories (in the reactors and in the fuel cycle). The major results are shown in Table XII.

TABLE XII. Annual Mass Balances (NPP of 60 GWe, 400 TWhe) at Equilibrium

	PWR – UOX Once-through	MIX[3]		CORAIL[4]		APA[4]		MOX-UE RMA
TRU in the core	--	Pu	Pu + MA	Pu	Pu + MA	Pu	Pu + MA	Pu
Fraction (%) in NPP	100	100	100	100	100	29	36	~ 30
Natural U need (t/year)	8271	7320	7576	7300	7700	6900	7000	6500
Enrichment need (MSWU/year)	6.3	5.5	5.7	5.5	5.9	5.1	5.2	4.8
Enrichment (%): ^{235}U	4.9[3]	4.5	4.7	4.8	5.0	3.3	3.9	1.9
Pu (tot)	--	2.1	2.8	7.8	10.3	12	12	9.7
MA	--	--	0.9	--	2.1	--	5	-
Fabrication / Reprocessing (tons/y)	817	--	--	742	742	} ~ 800	} ~ 790	230
UOX	--	817	817	346	346			590
Pu fuel (MIX/CORAIL/APA)								
To wastes[1]: TRU (Kg/y)	11767	3376	30.4	3537	45.3	2940	37.2	3610
Pu	10300	17.0	23.2	27.5	37.5	20	30	30
Np	738	662	1.0	606	1.2	580	1.2	640
Am	625	1800	2.6	2320	4.0	1740	3.0	2400
Cm	100	897	3.6	584	2.6	600	3.0	540
Radiotoxicity reduction factor	1	~ 3	~ 300	~ 3	~ 300	~ 3	~ 300	~ 3
Inventory in NPP (tons) [2]								
TRU ----------	94.1	246	394	338	508			
Pu ----------	82.4	220	300	314	422	Not	Not	Not
Np ----------	5.9	5.7	13	5	14	Available	Available	Available
Am ----------	5.0	12	34	14	40			
Cm ----------	0.8	8.4	47	5	32			

(1) Losses to wastes: Pu, MA 0.1%
(2) Cooling times: 5 + 2 years
(3) BU: 60 GWd/t
(4) BU: 45 GWd/t

3.2 THE IMPACT ON THE FUEL CYCLE

Detailed indications on the impact on the fuel cycle (e.g. at fuel fabrication) have also been obtained both at CEA and at ANL (Ref. 9). Some results for the CORAIL concept are summarized in Table XIII. These results are representative for all multirecycling concepts described previously. The increase in neutron dose is very significant. To appreciate the impact of the use of a thermal neutron spectrum, the results of one of the cases of (Pu + MA) multi-recycling in a PWR (namely the MIX concept), are compared with those obtained by the multi-recycling of (Pu + MA) in a standard MOX fuel of a fast reactor (in the present case, the European Fast Reactor, EFR). The results, comparable to those of Table XII, are given in Table XIV. The impacts on the fuel cycle (fuel fabrication) are still significant, but much reduced, due to the lower build-up of higher mass MA.

TABLE XIII
Impact on some fuel cycle parameters of the multiple recycle of all TRU in a LWR

Fuel cycle parameter(1)	Increase factor(2)	Comments(3)
Decay heat	~ 10	Cm-244 ~ 70 % ; Pu-238 ~ 30 %
Neutron source	~ 2000	Cf-252 ~ 90 % - If cooling time = 20 years, Cf-252 ~ 40 % ; the rest is due to Cm-244, Cf-250, Cm-250, Cm-246, Cm-248
Gamma source	~ 10	Am-241, Cm-244, Pu-238, Cf-252, Cm-243 are major contributors

(1) At fuel loading of 7th cycle, after fabrication and 5 years cooling.

(2) Increase in parameter, taken as the ratio to the corresponding parameter in a Pu-only MOX-LWR.

(3) When Pu only is multirecycled in MOX-LWR, Pu-238 is the major contributor to all parameters.

TABLE XIV. Annual Mass Balances (Fast vs Thermal System) at Equilibrium

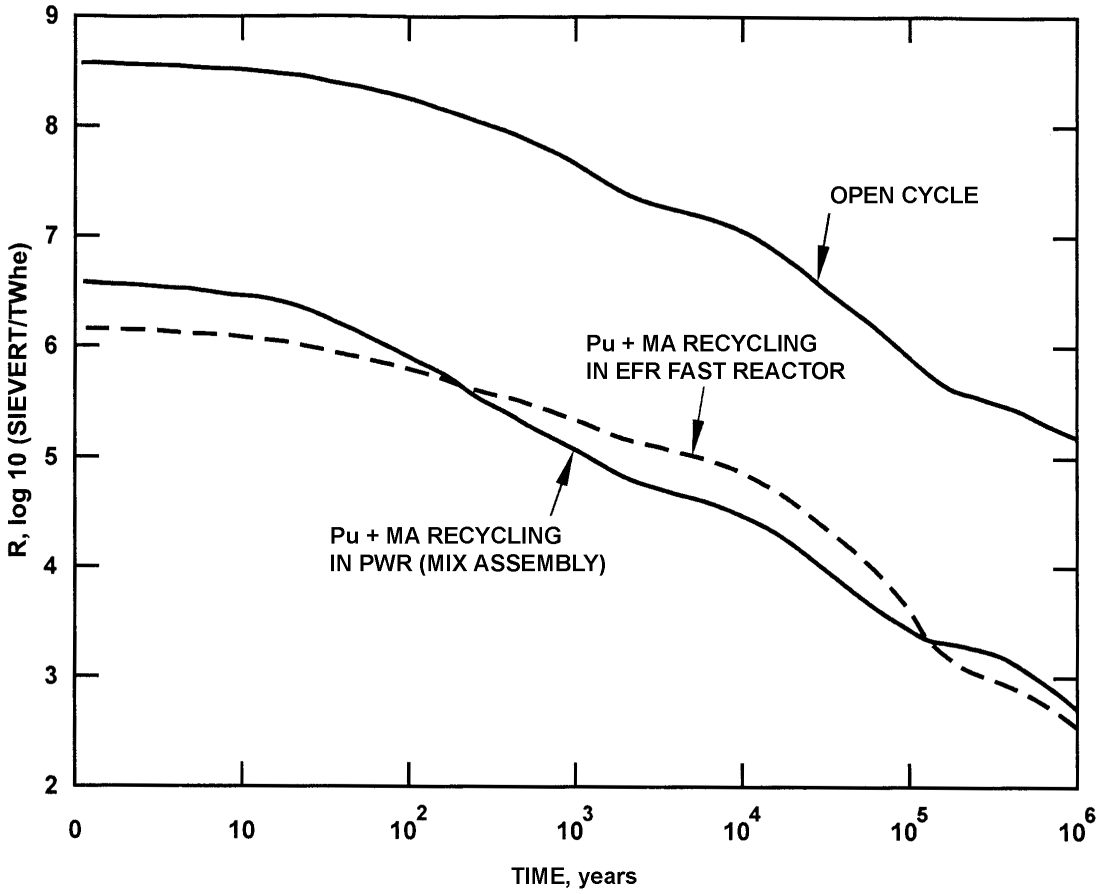
	PWR-UOX Once-through	CORAIL [4]		EFR (European Fast Reactor) Oxide Fuel (U, Pu) O_2	
TRU in the core	--	Pu	Pu + MA	Pu	Pu + MA
Fraction (%) in NPP	100	100	100	100	100
Natural U need (t/year)	8271	7300	7700	43.3	41.5
Enrichment need (MSWU/year)	6.3	5.5	5.9	--	--
Enrichment (%): ^{235}U	4.9 [3]	4.8	5.0	--	--
Pu (tot)	--	7.8	10.3	19.9	20.2
MA	--	--	2.1	--	1.2
Fabrication / Reprocessing (tons/y)					
UOX	817	742	742	--	--
Pu fuel (EFR/CORAIL)	--	346	346	336	357
To wastes[1]: TRU (Kg/y)	11767	3537	45.3	1759	60.5
Pu	10300	27.5	37.5	56.2	57.6
Np	738	606	1.2	171	0.3
Am	625	2320	4.0	1420	2.3
Cm	100	584	2.6	112	0.6
Radiotoxicity reduction factor	1	~3	~300	~12	~300
Inventory in NPP (tons) [2]					
TRU ----------	94.1	338	508	743	797
Pu ----------	82.4	314	422	730	753
Np ----------	5.9	5	14	1.3	4
Am ----------	5.0	14	40	11.2	32
Cm ----------	0.8	5	32	0.9	8

(1) Losses to wastes: Pu, MA 0.1%
(2) Cooling times: 5 + 2 years
(3) BU: 60 GWd/t
(4) BU: 45 GWd/t

The results of Tables XII-XIV show that:

- The radiotoxicity source term reduction is in all cases comparable, independently of the PWR multi-recycling option or even of the fast/thermal option. This means that, if "transmutation" is possible (i.e. if enough neutrons are available), the driving factor is the separation effectiveness. When the value for that parameter is assumed to be 0.1%, an average (over time) radiotoxicity reduction factor of approximately 300 has to be expected. A reduction of a factor ~ 3 is expected if only Pu (or Pu+Np) are multirecycled.
  Of course there are differences, e.g. between the fast and thermal spectrum systems, at different time scales, due to the presence of specific isotopes. An example is given in Figure 9.

FIGURE 9. Potential Radiotoxicity Source (R) Evolution with Time



- Recycling Pu and MA in a thermal spectrum, whatever the option, gives rise to a significant build-up of higher mass isotopes. The case of Cm is spectacular. In fact, practically in all scenarios, Cm does not reach a complete equilibrium (production = destruction). The Cm inventory (in the reactors and in the fuel cycle) is of the order of 30-50 t. In a fast spectrum, a lower build-up (by a factor 5-10) is shown.
  The higher mass isotope build-up in a thermal spectrum (which is due to the high capture cross-section and high capture-to-fission cross-section ratios below ~ 100 eV) has a significant impact on some crucial operations of the fuel cycle, like fuel fabrication. This effect and its impact are shown in the results of Table XIII. The high neutron doses in the thermal PWR spectrum are essentially due to the build up of Cf-252 see Fig. 10. Even modest amounts of that isotope, which has a very high neutron emission yield by spontaneous fission (~ 10^{12} neutrons/g.s), give rise to very significant neutron doses.

Specific measures have to be taken at fuel fabrication to improve the standard remote handling of Pu-based fuels. The impact of Pu only multirecycling, is compatible with present installation (like MELOX) performances. The presence of Cm (and, at a lesser degree, of Am) in the fuel, is the real difficulty.

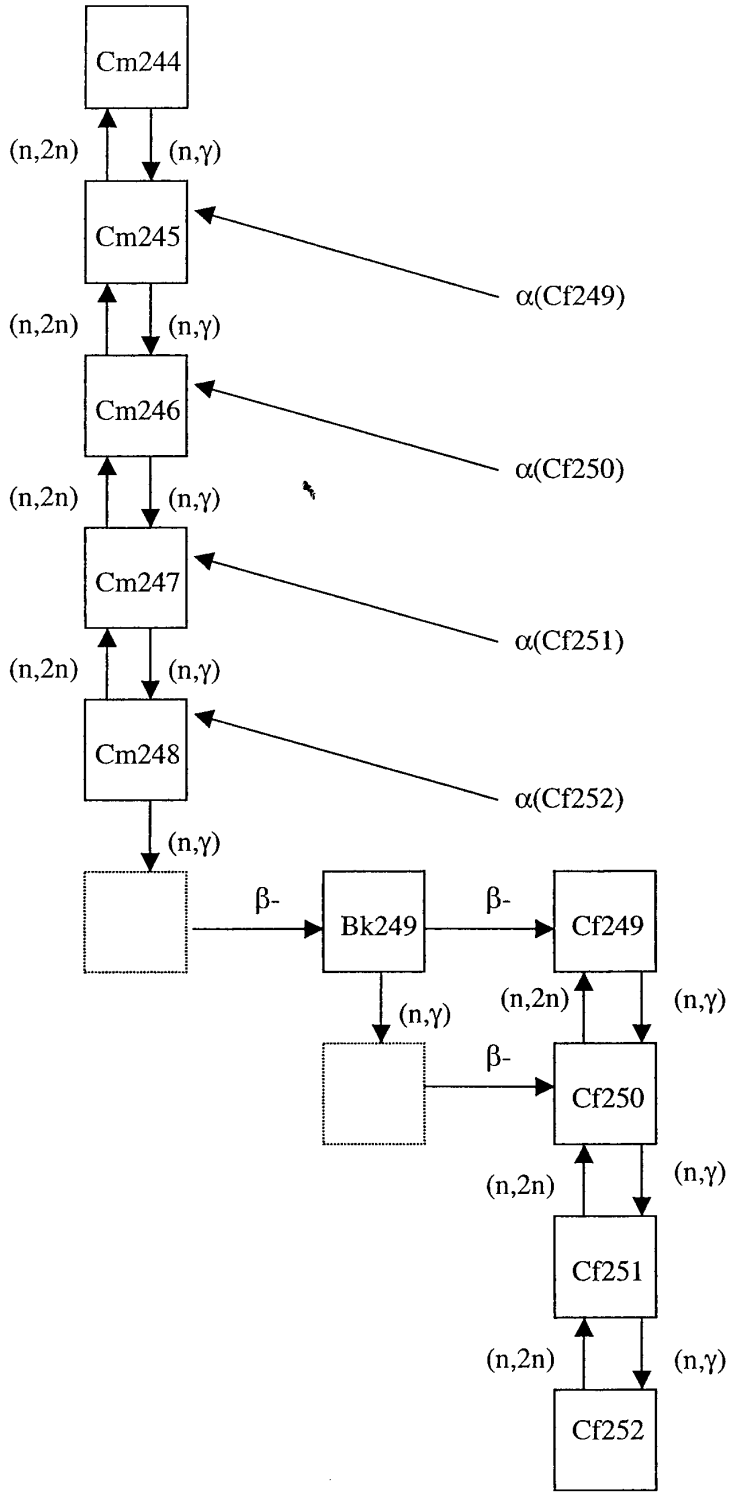


FIGURE 10
Build-up of Cf252 under irradiation

- The mass inventories in the NPP show that the remarkable benefits of Pu and MA recycling in terms of reduction of the masses of high level radioactivity wastes sent to a repository, are obtained with the increase of transuranium elements in the fuel cycle (reactor + out-of-core fuel cycle). With respect to this last point, the split between in-core/out-of-core is, e.g. for Pu, the following: PWR-UOX 30.4/52; MIX (Pu) 100/120; MIX (Pu + MA) 137/163; CORAIL (Pu) 121/193; CORAIL (Pu + MA) 162/260.

- The use of thermal reactors does not decrease the total inventory (with respect to the use of fast spectrum cores), but increases very significantly the Cm inventory, as noted above.

- It is obvious that only the use of fast spectrum systems reduces drastically the need for natural uranium (reducing in that way the doses associated to mining and processing).

3.3 SELECTIVE RECYCLING OF TRU IN LWR'S AND THE PROBLEM OF CURIUM

Studies performed at ANL [Ref. 9], have consolidated and enlarged the scope of the original Pu or TRU recycling studies in FRANCE on the CORAIL concept.

In particular, crucial results have been obtained for selective TRU-multirecycling strategies.

These studies have two objectives:

1) Assess the proliferation resistance characteristics of a Pu+Np (or Pu+Am) homogeneous recycling and possible improvements with respect to the case of Pu-only multirecycling.

2) Assess the potential reduction of the impact of a full TRU recycling on the fuel cycle, as indicated in the previous paragraph. In fact, the feasibility of a full, indefinite multirecycling of TRU, looks difficult in terms of the extra costs induced by the measures to be taken in particular at fuel fabrication.

Some of the results obtained in [9] are summarized in Table XV. The results, consistent with those of table XIII, are relative to an equilibrium state. The selective recycling of Pu+Np or Pu+Np+Am strongly reduces the burden on the fuel cycle and has an impact on the repository criteria; even if the decay heat increase when multirecycling Pu+Np+Am is still significant as shown in Table XV.

TABLE XV : Comparison of Equilibrium State Fuel Cycle Parameters

		CORAIL-Pu taken as reference	**CORAIL TRU**	**CORAIL Pu+Np**	**CORAIL Pu+Np+Am**
Decay Heat	Fabrication	1	20	1.9	5.9
	Charge	1	18	1.9	5.8
	Discharge	1	1	1	1
	After 5 years cooling	1	3	1.1	2.2
Neutron Source	Fabrication	1	6375	1.5	3.8
	Charge	1	3.8×10^4	1.5	3.8
	Discharge	1	947	0.98	2.6
	After 5 years cooling	1	395	0.94	2.4
Gamma Source	Fabrication	1	41	1.9	13.7
	Charge	1	22	1.6	8.5
	Discharge	1	0.97	1	0.98
	After 5 years cooling	1	0.95	0.99	0.95

In the case of the multirecycling of Cm, as previously indicated, the Cm isotope equilibrium is not reached after 10 cycles, in particular for Cm-248, which is the gateway towards Cf-252 (see figure 11), despite the fact that Cm-244 is close to equilibrium (see figure 12), Cf and Cm isotopes properties are summarized in Table XVI. The Cf evolution with cycle is given in figure 13. Both Cm-248 and Cf-252 will eventually reach equilibrium but much later on (e.g. in the case of Cm-248 as shown in Fig.11, the equilibrium value corresponds to approximately 0.1 Kg/assembly).

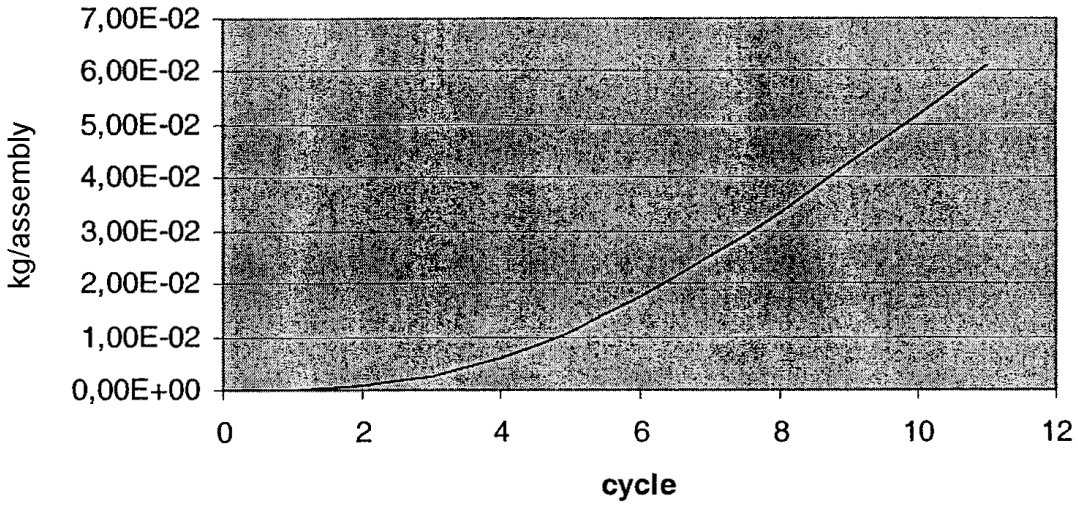


FIGURE 11. Evolution of Cm-248 as a function of LWR recycle

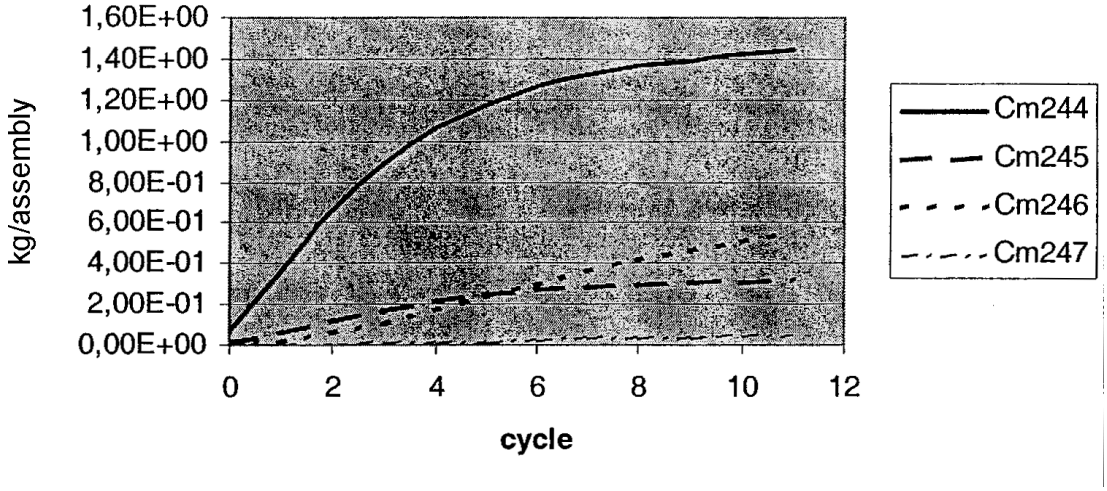


FIGURE 12. Evolution of curium isotopes as a function of LWR recycle

TABLE XVI
Properties of Cm and Cf isotopes

Nuclide	Half-life	Neutrons/g/s		Eγ (keV)	γ rays	nSv/Bq (ICFR-72)
		Spontaneous Fission (SF)	(α,n)			
Cm242	163 d	$1.72\ 10^{7}$	$4.18\ 10^{6}$	1.4		12
Cm243	30.0 y		$6.09\ 10^{4}$	133.2	104 keV (24 %) 228 keV (11 %) 278 keV (14 %)	150
Cm244	18.1 y	$1.01\ 10^{7}$	$8.84\ 10^{4}$	1.3		120
Cm245	$8.50\ 10^{3}$ y			93.8	104 keV (30 %) 100 keV (18 %)	210
Cm246	$4.73\ 10^{3}$ y	$\approx 7\ 10^{6}$	<< SF	3.0		210
Cm247	$1.60\ 10^{7}$ y			302.8	403 keV (69 %)	190
Cm248	$3.40\ 10^{5}$ y	$\approx 3\ 10^{7}$	<< SF	579.1	579 keV (100 %)	770
Cf249	351 y		n.a.	329.2		350
Cf250	13.1 y	$\approx 8\ 10^{9}$	<< SF	6.3		160
Cf251	898 y		n.a.	120.3		360
Cf252	2.64 y	$\approx 10^{12}$	<< SF	217.4		90

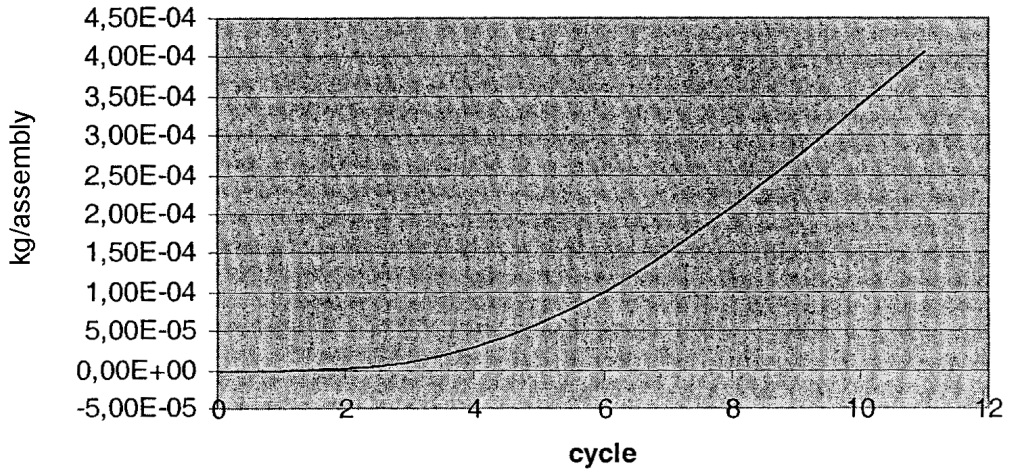


FIGURE 13. Evolution of Cf-252 as a function of LWR recycle

However, to avoid these difficulties, it has to be noticed that, letting Cm+Am or even Cm, go to the wastes, dramatically reduces the gain in terms of, e.g., radiotoxicity source (the gain of a factor of ~ 300 indicated in paragraph 3.1, is reduced to approximately ~ 10, in the case of Pu+Np+Am multirecycling), or decay heat in the repository. Moreover, a clear strategy has to be defined for Cm. One can imagine a scenario of intermediate storage of Cm, to let it decay (mostly in Pu-240) and sending it back (mainly as Pu), after ~ 100 years, to the fuel cycle. The extra amount of even Pu isotopes could eventually be acceptable if mixed to recycled Pu, but only in future fast reactor cores.

This ideal strategy has still to be worked-out in detail, since the intermediate storage of Cm is characterized by high specific heat values (cooling problems) and by criticality-safety issues. Moreover, the form of stored Cm (e.g. its blending with a matrix like U from reprocessing) should be specified, in order to envisage a process for its re-injection in the fuel cycle (under the form of separated Pu-240).

Some data related to the Curium decay are given in figures 14-15 and table XVII. The quantities provided are the masses, decay heat, radiotoxicity and neutron source arising from the Curium in the spent fuel of the CORAIL concept with Pu+Np+Am recycle (per metric ton of the fuel).

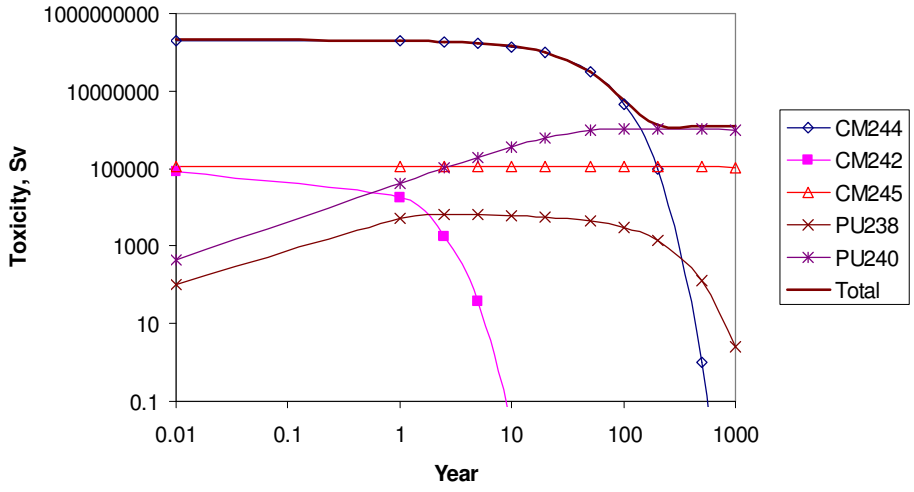


FIGURE 14
Radiotoxicity comparison of curium decay contributors

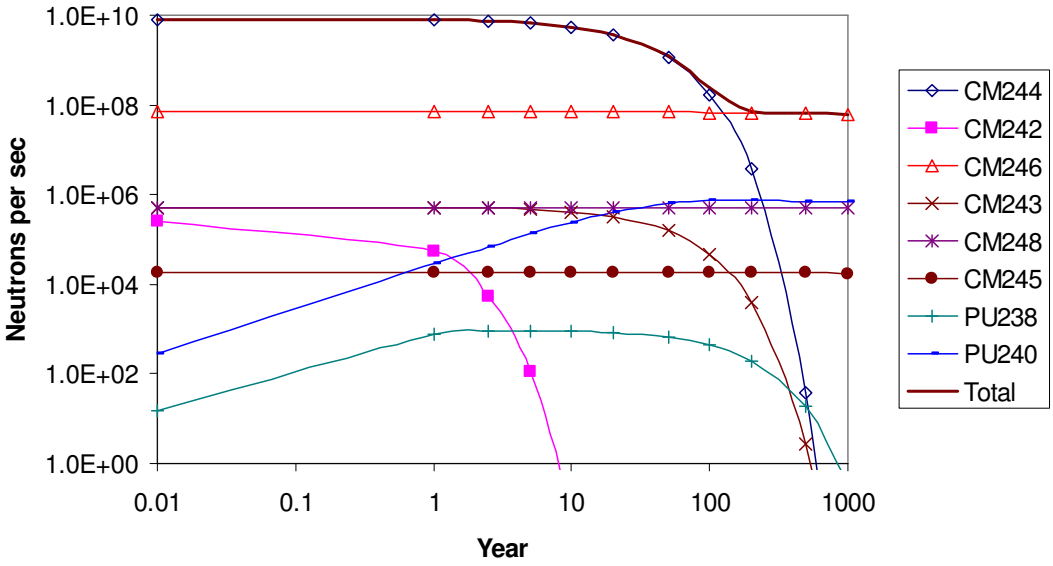


FIGURE 15
Comparison of neutron emission rate of curium decay contributors
(Both of spontaneous and (α,n) sources)

TABLE XVII
Leading contributors [a)]

years		0	0.01	1	2.5	5	10	20	50	100	200	500	1000
Mass, g	CM244	707.30	707.00	680.70	642.70	584.10	482.40	329.00	104.30	15.39	0.34	0.00	0.00
	CM245	113.50	113.50	113.50	113.50	113.50	113.40	113.30	113.00	112.60	111.70	109.00	104.60
	PU240	0.00	0.27	26.12	63.47	121.10	221.10	371.70	591.00	675.10	682.70	661.60	627.40
	U234	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.02	0.03	0.05	0.06	0.06
	U236	0.00	0.00	0.00	0.01	0.03	0.12	0.44	2.01	5.38	12.49	33.52	67.11
	Total	836.30	836.29	836.29	836.32	836.33	836.35	836.21	836.34	836.37	836.34	836.26	836.33
Decay heat, Watt	CM244	2002.00	2002.00	1927.00	1820.00	1654.00	1366.00	931.30	295.40	43.58	0.95	0.00	0.00
	CM243	14.31	14.31	13.97	13.47	12.67	11.22	8.80	4.24	1.26	0.11	0.00	0.00
	CM242	7.24	7.13	1.54	0.15	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	PU238	0.00	0.00	0.03	0.03	0.03	0.03	0.03	0.02	0.02	0.01	0.00	0.00
	PU240	0.00	0.00	0.19	0.45	0.86	1.57	2.64	4.20	4.79	4.85	4.70	4.45
	Total	2024.28	2024.17	1943.44	1834.83	1668.29	1379.55	943.50	304.62	50.45	6.80	5.74	5.61
Radio-toxicity, Sv	CM244	2.055E+8	2.054E+8	1.978E+8	1.867E+8	1.697E+8	1.401E+8	9.558E+7	3.031E+7	4.472E+6	9.733E+4	1.003E+0	4.899E-9
	CM242	8.729E+4	8.596E+4	1.850E+4	1.805E+3	3.734E+1	1.598E-2	2.925E-9	0	0	0	0	0
	CM245	1.154E+5	1.154E+5	1.154E+5	1.154E+5	1.154E+5	1.153E+5	1.152E+5	1.150E+5	1.145E+5	1.135E+5	1.108E+5	1.064E+5
	PU238	0	1.026E+2	5.228E+3	6.432E+3	6.439E+3	6.192E+3	5.722E+3	4.514E+3	3.042E+3	1.381E+3	1.291E+2	2.486E+0
	PU240	0	4.267E+2	4.186E+4	1.017E+5	1.941E+5	3.543E+5	5.957E+5	9.469E+5	1.082E+6	1.094E+6	1.060E+6	1.005E+6
	Total	2.074E+8	2.073E+8	1.996E+8	1.886E+8	1.716E+8	1.420E+8	9.738E+7	3.191E+7	5.857E+6	1.368E+6	1.250E+6	1.215E+6
Neutron source n/sec	CM244	7.93E+09	7.93E+09	7.63E+09	7.21E+09	6.55E+09	5.41E+09	3.69E+09	1.17E+09	1.73E+08	3.76E+06	3.87E+01	1.89E-07
	CM242	2.64E+05	2.60E+05	5.59E+04	5.46E+03	1.13E+02	4.83E-02	8.84E-09	5.42E-29	0	0	0	0
	CM246	6.90E+07	6.90E+07	6.90E+07	6.90E+07	6.90E+07	6.89E+07	6.88E+07	6.85E+07	6.80E+07	6.70E+07	6.42E+07	5.96E+07
	CM243	5.19E+05	5.19E+05	5.07E+05	4.89E+05	4.60E+05	4.07E+05	3.19E+05	1.54E+05	4.56E+04	4.01E+03	2.72E+00	1.42E-05
	CM248	5.01E+05	5.01E+05	5.01E+05	5.01E+05	5.01E+05	5.01E+05	5.01E+05	5.01E+05	5.01E+05	5.01E+05	5.00E+05	5.00E+05
	CM245	1.86E+04	1.86E+04	1.86E+04	1.86E+04	1.86E+04	1.86E+04	1.86E+04	1.86E+04	1.85E+04	1.83E+04	1.79E+04	1.72E+04
	PU238	0	1.47E+01	7.47E+02	9.19E+02	9.20E+02	8.85E+02	8.17E+02	6.45E+02	4.35E+02	1.97E+02	1.84E+01	3.55E-01
	PU240	0	2.88E+02	2.83E+04	6.88E+04	1.31E+05	2.39E+05	4.03E+05	6.40E+05	7.31E+05	7.40E+05	7.17E+05	6.80E+05
	Total	8.00E+09	8.00E+09	7.70E+09	7.28E+09	6.62E+09	5.48E+09	3.76E+09	1.24E+09	2.42E+08	7.21E+07	6.54E+07	6.08E+07

a) Summation of all leading contributors is about 99% total value.

An alternative way out could be to envisage a scenario where, in the so-called Tier-1, TRU are recycled only twice, with longer cooling times after discharge (7-10 years), in order to prepare, and favor, a successive introduction of fast reactors, which will allow to continue the multirecycling. In that case, the limited recycling in LWRs will act as a "delay line" towards the successive introduction of FRs which will be able to implement in a sustainable way the full-TRU multirecycling. A further potential advantage of such strategy will be to develop a unique fuel cycle option, based on the non-separation of TRU, which is probably the most advantageous in terms of non-proliferation resistance.

4. The heterogeneous recycling of Americium

In the previous paragraphs, we have dealt with TRU recycling and transmutation in LWRs, according to the so-called "homogeneous" mode in standard LWRs. One recycle of Pu has already been demonstrated at the industrial level and Pu-only multi-recycling is certainly feasible, and LWRs offer several credible possibilities to manage Pu stocks in a flexible way. The introduction of MA, even if in principle possible, gives rise to potential practical difficulties and economic penalties, in particular in the fuel handling.

There have been studies (and some are still underway) devoted to the consumption of Pu and MA using new fuel forms to be loaded in standard LWRs. The use of new fuel forms implies that these options are less "short term" options with respect to that described in Section 3. In reviewing these more innovative options, we will give more a survey than a systematic comparison, in view of the more exploratory nature of the studies already performed or ongoing.

In the present section, we first review the so-called "heterogeneous" recycling, namely the use of MA targets, separated from the main fuel, to be irradiated in specific assemblies or in specific positions inside one assembly. The advantage of this strategy is clearly to separate the MA cycle (target fabrication, their potential reprocessing etc.) from the main fuel cycle.

However, several questions have to be answered:

- which is the optimum neutron spectrum and flux level,
- should Am be separated from Cm,
- once-through irradiation or multirecycling of the targets,
- what type of matrix as support to Am (or Am+Cm),
- are there technological limits (power evolution in the target, fluence and cladding material damage).

4.1 THE OPTIMUM SPECTRUM AND FLUX LEVEL

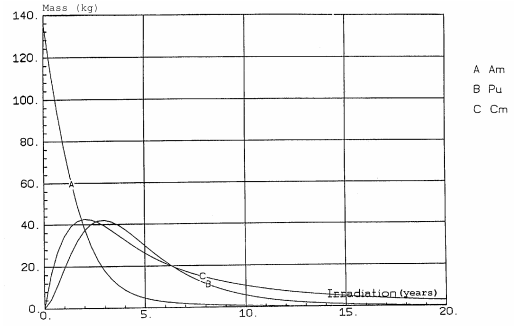
The flux transmutation of Am isotopes (down to fission products) is better achieved in a fast neutron spectrum, in terms of neutron economy (see paragraph 7). However, the high reaction rate values of Am isotopes in a thermal spectrum gives rise to a faster transmutation process, of course at the expense of the neutron balance, as we will see later on.

In an ideal application to Am-241, Reference [10] quotes ~ 30 years to fission half of the initial Am-241 and reaction products in a fast spectrum (flux level: 10^{15} n/cm^2.sec) and ~ 25 years to fission 95 % of the initial Am-241 and reaction products in a PWR spectrum (flux level : 3 x 10^{14} n/cm^2.sec).

To find out an optimum compromise, it was proposed to envisage the irradiation of Am (or Am+Cm) targets at the periphery of a fast reactor, where the flux level is still high ($\approx 10^{15}$ n/cm.sec), but in special subassemblies, where a moderator material with low capture (e.g. CaH_2) is put around the targets to thermalise the incoming neutrons. In this way the transmutation rate is maximized (high ϕ, high σ) and the fast neutron reactor environment insures a suitable neutron balance.

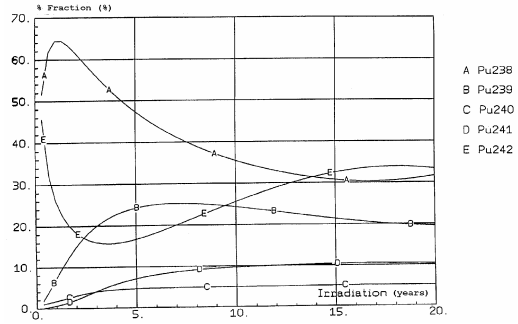
However, the irradiation time needed to reach 95 % cumulative fissions is still very long, as illustrated in fig. 16-18, where an Am on an inert matrix target is irradiated according to the scheme indicated above. The build-up of Pu and Cm isotopes is shown. Moreover very high doses are reached, and the expected dpa values above 200-300 require special attention.

FIGURE 16



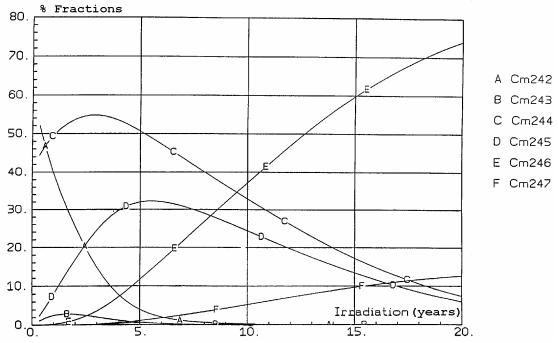
Irradiation of Am in a Moderated S/A at the periphery of a standard fast reactor - Am elimination and Pu, Cm build-up

FIGURE 17



Irradiation of Am in a Moderated S/A - Pu isotopes breakdown

FIGURE 18



Irradiation of Am in a Moderated S/A - Cm isotopes breakdown

4.2 MA TARGETS IN LWRS

In any case, the compromise solution described in 4.1, is only applicable in a fast reactor. However, the use of targets in standard LWRs has also been envisaged. The underlying rationale is to limit to a minimum the burden on the fuel cycle and to envisage a once-through strategy. In this case, the parameter which defines the transmutation performance, is no more the recovery effectiveness at reprocessing (homogeneous recycling), but the cumulative fission rate which can be reached under realistic conditions.

In the TIGRE concept [11] Am targets are inserted in each guide tube of a standard PWR assembly (see Fig. 19).

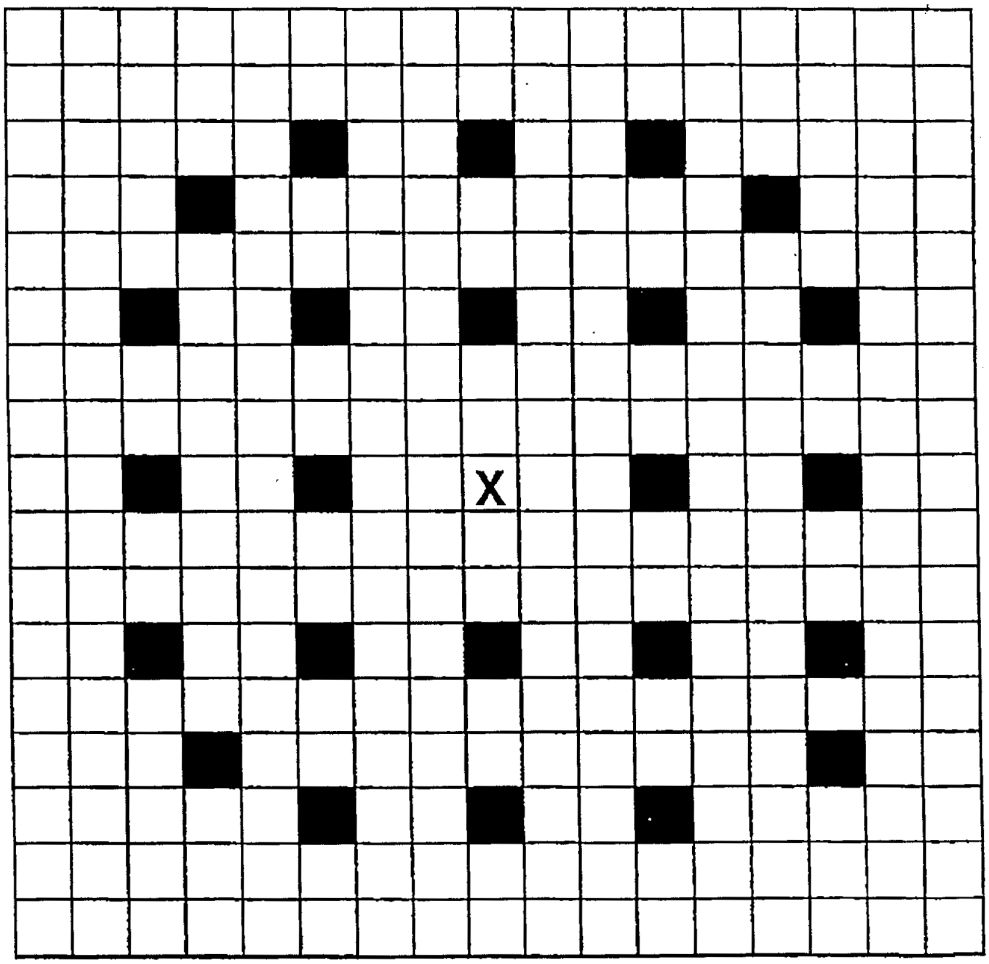


FIGURE 19
PWR-UOX assembly with MA targets

Standard UO_2 pin

Guide tube with targets ⟹ { 24 targets for each assembly; Each target : 30 % AmO_2 + 70 % Al_2O_3

X Instrumentation tube

First, multirecycling of the target has been considered.

For targets with Am only, an equilibrium state (i.e. productions=destruction in the reactor) can be reached, but almost all PWRs have to be equipped with TIGRE assemblies (~ 70 for each PWR). This number is imposed to limit the overenrichment in ^{235}U (PWR-UOX) or Pu content (PWR-MOX).

The linear power in the target at the end of irradiation is 130 W/cm, which can be a problem for cooling, according to the design of the guide-tubes in different PWRs.

If (Am+Cm) is put in the targets, the number of TIGRE assemblies required increases beyond the constraint of allowable assemblies (constraint on enrichment) and an equilibrium state cannot be found.

Limited gains on the radiotoxicity in a repository are obtained with the “Am - only” strategy (e.g. a further reduction of a factor of 2÷3 on radiotoxicity with respect to the “Pu-only” recycling scenario). However, the recent studies by R. Wigeland et al. at ANL (Ref.20), indicated the relevance of reducing the heat loading coming from Am-241 in the specific case of Yucca Mountain and future studies should take that aspect into consideration.

The monorecycling of targets in the TIGRE assembly has also been attempted, both for “Am-only” and for Am+Cm targets.

The major results are as follows (“Am-only” targets) :

- For each PWR-UOX, 66 assemblies with 24 targets are introduced.
- To reach 90 % cumulative fission rate, 25 years irradiation are needed. To reach 97 %, 40 years are needed (of course, shorter irradiation times may be adequate if the relevant parameter is the heat load in a repository due to just Am-241, and not the total mass MA).
- Since to reach these high fission rates one has to decrease the amount of Am in each target, the transmuted masses are modest (2.5 to 1.5 kg/TWhe). To increase that, there is a need to overenrich in ^{235}U. In any case, the power variation during irradiation is very significant.

An alternative approach has been studied (Ref. 12): a “dedicated” assembly (ANDIAMO), with MA-based pins on a fissile support.

The loading of the targets in heavy atoms is 60 % by mass in an inert matrix. The composition is: ~ 50 % Pu, 50 % MA. The following results were obtained:

- The power and reactivity variations can be optimized and the “dedicated” assembly can be made “transparent” with respect to the surrounding PWR assembly environment.
- One can reach a sizable destruction rate (~ 5.8 kg/TWhe of Am), which allows an equilibrium state in a reactor park.

However, irradiation times are still long to be realistic (~ 30÷40 years to reach 96 % fissions).

More recently [Ref. 6] in the frame of the MOX-UE studies, the heterogeneous recycling of Am has been again considered.

The Am is recycled in UO_2-AMO_2 targets which are located in 16 of the 36 water rods. The ratio Am/Am+U is 14.25 % in the targets so that averaged over the entire assembly the Am content does not exceed 1 % for safety reasons (void coefficient).

In this case, preserving the fuel cycle length makes it necessary to increase the MOX-UE U-235 enrichment by about 2 %.

After the 3x505 EFPD irradiation, about 57 % of the initial Am has been transmuted in the targets, corresponding to about 11 kg/TWhe. The total net Am production in the assembly is then only about 2 kg/TWhe against about 13 kg/TWhe when it is not recycled. On the other hand the Pu consumption, which is about 70 kg/TWhe when Am is not recycled, is now only about 50 kg/TWhe.

One possibility allowing to recycle the totality of the Am, could be to load Am targets in all MOX-UE assemblies plus in a certain number of UOX assemblies. To equal Pu-Am production and consumption, a preliminary estimation gives : ~ 40 % MOX-UE with Am targets + ~10 % UOX with Am targets + ~ 50 % standard UOX. Recent preliminary calculations performed at ANL, confirm that a five years irradiation of 100% Am targets reduces the initial mass by 22% and that the residual material has 8% Am-241, 51% Pu-238, 10% Pu-242, and 74% total Pu by weight.

For all these cases, the impact on the heat load and its evolution in time in a repository should be verified, and a credible strategy (storage, separations, processing etc.) for the management of the residual products (see section 3.3 and next section 4.3) should be explored.

4.3 THE PROBLEM OF Cm

In the case of the heterogeneous recycling, one has to deal with the problem already encountered for the homogeneous recycling of MA, i.e. how to deal with Cm. As we have seen, there have been studies both for Am-only and Cm+Am targets. Once more, the burden (essentially in the fuel cycles at the fabrication stop) to keep Cm in the target is the condition to further reduce the radiotoxicity of the wastes sent to the repository.

However, since the monorecycling of targets is already definitely less effective than the homogeneous multirecycling, the effect of Cm separation has less impact in the overall transmutation performance.

Comparative results are not available for LWR - based scenarios. However some hints can be obtained from scenario studies performed at CEA for fast reactors. Three scenarios are compared, i.e. a FR-only power park a) with homogeneous recycling of all TRU b) homogeneous recycling of Pu and Np and monorecycling of Am+Cm targets in "moderated" subassemblies at the core periphery c) same but Am-only targets and Cm sent to the repository.

The results both in terms of radiotoxicity reduction and mass reduction are given in Table XVIII and Table XIX respectively, and in terms of decay heat and neutron source increase at fabrication in Table XX.

TABLE XVIII
Waste Mass (kg/y) sent to the Repository: Reduction due to heterogeneous monorecycling for a 400 TWhe power park

Scenario	Mass to the wastes					
	Np	Pu	Am	Cm	Total	Reduction factor
Reference: UOX-PWR open cycle	738	10 300	625	104	11 767	1
Homogeneous recycling of all TRU in FR	0.31	57.6	2.3	0.63	60.8	195
Homogeneous recycling of Pu+Np in FR Monorecycling[(a)] of Am+Cm targets at core periphery	0.4	123	16	65	204.4	58
Homogeneous recycling of Pu+Np in FR Monorecycling[(a)] of Am targets at core periphery	0.7	106	14.3	181	302	38

(a) Cumulative fission rate: 90 %

TABLE XIX

Scenario	Radiotoxicity reduction factor due to heterogeneous recycling of Am+Cm			
	10^3 y	10^4 y	10^5 y	10^6 y
Reference: UOX-PWR open cycle	1	1	1	1
Homogeneous recycling of all TRU in FR	210	150	218	445
Homogeneous recycling of Pu+Np in FR Monorecycling[(a)] of Am+Cm targets at core periphery	70	50	33	45
Homogeneous recycling of Pu+Np in FR. Monorecycling[(a)] of Am targets at the periphery of the core	35	25	23	30

(a) Cumulative fission rate: 90 %.

TABLE XX
Consequences on the fuel fabrication of the heterogeneous monorecycling of MA targets

Fuel of scenario	Homog. recycling of TRU in a	Monorecycling of Am+Cm targets
Decay heat	X 7[(a)]	X 36
Neutron source	X 150	X 500

(a) Reference: FR fuel without MA. (X 7: increase by a factor of 7).

These results illustrate the arguments given above, even if they should not be directly applied to a LWR case.

4.4 THE MATRIX SUPPORT FOR THE MA TARGETS

Studies (also experimental) have addressed both inert matrix concepts (e.g. using MgO or Al_2MgO_4) or UO_2-based matrix concepts. As far as practical applications, the UO_2 matrix is attractive (well known properties under irradiation, potential for easy reprocessing if needed etc.), and the extra Pu production under irradiation, is significantly less than the Pu production due to the Am irradiation.

5. The Inert Matrix Fuel (IMF) concepts

This section is devoted to innovative concepts based on new fuel forms to be introduced in standard LWRs. The exploratory nature of most of the studies performed up to now, and the limited experience on the actual fabrication processes, basic properties and performance under irradiation of the proposed fuels, does not allow a systematic intercomparison, and a survey approach for a few significant examples has been taken.

5.1 PU IN AN INERT MATRIX FUEL

The Inert Matrix Fuel (IMF) concepts have been introduced and widely studied in Europe and in Japan, at the beginning essentially as concepts devoted to the reduction of Pu stocks. The leading idea is to provide a U-free support to Pu, to burn it in a once-through mode in LWRs. The form of the spent fuel should be compatible with its final disposal, or further burning in a dedicated transmuter.

From the physics point of view, a homogeneous mixture of plutonium and a suitably chosen burnable poison can have a relatively flat reactivity variation with burn-up, allowing for a significant reduction of the total initial plutonium (up to ~ 60 % in the case of reactor-grade plutonium) with the elimination of most of the fissile Pu, up to 90 - 95 % [13]. Reactivity coefficients can be kept close to acceptable values, despite a reduced Doppler, βeff and boron effectiveness [13]. However, most of the analysis performed up to now, should be confirmed with 3-dimensional dynamic calculations for well defined configurations.

The technically most advanced proposal for an LWR IMF for plutonium burning is (Ref. 14) a once-through mode concept, viz. that of a homogeneous solid solution of PuO_2 in stabilized ZrO_2 containing erbium as burnable poison. An alternative (fertile) dopant which has been suggested is thorium. The radiation stability of the ZrO_2-based solid solution, its low solubility under repository conditions, and its straightforward fabrication process (compatible with current MOX technology, e.g. the possibility of building a new line in an existing MOX plant) are important advantages of this concept. However, the lower thermal conductivity, and the concomitant higher fuel temperature resulting in a higher fission gas release and smaller margin to melting compared to UO_2, represents a major uncertainty that needs to be quantified accurately. Irradiation tests are required for this. Two relevant experiments - the comparatory irradiation of IMF and MOX rod specimens in the Halden reactor and the OTTO Pu-incineration experiment in the HFR at Petten are underway and first results are being analyzed. Important information about the in-pile behavior of ZrO_2-based fuels, i.e. their thermomechanical properties as function of burn-up, can thus be expected by 2005-6.

As a consequence of the deep Pu burning, the amount of MA built-up (in particular Cm) is significant, up to ~ 5 % of the initial Pu. An example is given in Table XXI taken from Ref. 13. The MA build-up is comparable to that obtained with a standard MOX core. However, the minor actinide contribution to repository criteria (e.g. dose and heat load) may not be sufficiently reduced if the spent fuel is directly disposed (Ref.20). Thus, recycle is likely required to achieve significant repository benefits.

TABLE XXI
BOL and EOL Concentrations (Relative to Initial Pu mass) of Pu Isotopes and Minor Actinides (Assembly Averages) for the MOX and U-free Cores
(RG : Reactor Grade - WG : Weapon Grade)

	RG-MOX		WG-MOX		RG-Ufree		WG-Ufree	
	BOL (%)	EOL (%)	BOL (%)	EOL (%)	BOL (%)	EOL (%)	BOL (%)	EOL (%)
Depletion	0.0	1300	0.0	1300	0.0	1300	0.0	1350
^{238}Pu	1.3	1.3	0.0	0.6	2.7	2.0	0.0	0.2
^{239}Pu	60.8	23.9	93.4	26.6	54.5	6.5	93.4	6.0
^{240}Pu	24.7	21.7	6.0	20.2	22.8	15.1	6.0	11.6
^{241}Pu	8.7	12.3	0.6	11.1	11.7	9.1	0.6	6.6
^{242}Pu	4.5	7.6	0.0	4.7	8.3	10.2	0.0	3.4
Total Pu	100.0	66.8	100.0	63.2	100.0	42.9	100.0	27.8
^{237}Np	0.0	0.41	0.0	0.65	0.0	0.00	0.0	0.0
^{241}Am	0.0	0.84	0.0	0.53	0.0	0.71	0.0	0.34
^{243}Am	0.0	2.04	0.0	1.09	0.0	2.22	0.0	0.76
^{242}Cm	0.0	0.21	0.0	0.17	0.0	0.23	0.0	0.15
^{244}Cm	0.0	1.23	0.0	0.47	0.0	1.55	0.0	0.37
^{245}Cm	0.0	0.14	0.0	0.04	0.0	0.18	0.0	0.03
Total minor actinides	0.0	5.02	0.0	2.95	0.0	4.99	0.0	1.65

Recently, an interesting application of the IMF concept has been made to the CORAIL concept [7].

The principle is very simple : in the original CORAIL assembly layout (i.e. 180 UOX rods, 84 MOX rods), the MOX rods are replaced by IMF rods (CeO_2-PuO_2).

The feasibility of the concept has been shown, at least from the point of view of the reactor parameters.

The need of U-235 enrichment is somewhat lower (from 5.4 % to 5 %), the Pu consumption goes from -3 kg/TWhe (standard CORAIL concept), corresponding to ~ 5.5 % of the initial Plutonium, to - 20 kg/TWhe, i.e. ~ 40 % of the initial Plutonium. The build-up of Am and Cm stays approximately the same.

For the CORAIL-IMF concept, ~ 50 % of the reactors should be loaded with the CORAIL-IMF assemblies, in order to stabilize the Pu inventory.

5.2 PU AND MA IN AN INERT MATRIX FUEL

At MIT several concepts have been developed [Ref. 15] based on IMF in order to burn/manage all TRU. As far as TRU burning in a once-through mode, both homogeneous and heterogeneous assembly concepts have been studied. The fuel is made with $MgAl_2O_4$ as primary inert matrix host material, and the TRU are dispersed as micro-size particles in that

inert matrix. Yttria stabilized Zirconia (YSZ) was chosen to be a part of the micro-spheres composition in order to enhance the irradiation and mechanical stability of the fuel particles.

In the homogeneous option, a standard PWR assembly layout has been adopted, since parametric studies did not show significant benefits of changing the moderator-to-fuel ratio (e.g. by changing the lattice pitch). In the heterogeneous option, Pu is concentrated in one type of fuel assembly, while all of the MA are concentrated in the second type of assembly. The ratio Pu/MA of the initial loading, is the one of a spent UOX fuel (4.2 % U-235 enrichment, 50 GWd/t BU), after 10 year cooling. At discharge, ~ 50 % of the initial TRU loading has been burnt, with the expected degradation of the Pu vector, very similar to that observed in the case of the PSI concept given in table XXI. Approximately 60% of the initial Pu is burned, as in the case of the PSI concept.

The heterogeneous loading does not bring any significant improvement with respect to the homogeneous loading in terms of Pu destruction, MA destruction being less favorable in the heterogeneous loading.

As far as core performances, the results obtained do not indicate significant problems related to the reactivity feedback coefficients. However, much smaller soluble boron worth and effective delayed neutron fraction can impose major limitations on the feasibility of a PWR core with 100 % loading of TRU in an inert matrix fuel. This conclusion is rather more pessimistic than the conclusion (limited to Pu in an IMF) of the PSI study, quoted in section 5.1.

The MIT study offers also a different solution, i.e. a Combined-Non-fertile and Uranium (CONFU) assembly to be loaded in a standard PWR core, in order to stabilize production and destruction of TRU.

The assembly is a standard PWR assembly, where ~ 60 UO_2 pins are replaced with IMF fuel, bearing Pu and MA actinides. The results show the potential of the CONFU concept, when multirecycling is applied. The TRU isotopic vector composition is stabilized for Pu and Am. Cm isotope composition does not saturate and Cm is still building up after 5 recyclings.

These results should be compared with a similar, and somewhat more detailed, study performed with the APA concept. In fact an attempt has been made to estimate the potential of (Am+Cm) incineration in APA PWR with a higher moderation ratio, and with Plutonium for the first cycle obtained from a standard UOX (1st generation Plutonium). It seems that the mixing of 8 Kg/ass. of (Am+Cm) (that corresponds to 2.8 % in mass) with 33 Kg/ass. of Pu permits to reach the equilibrium between production and consumption. In case of *(Pu+Am+Cm) multi-recycling in APA PWR* and at equilibrium, the total consumption of Pu is 53 Kg/TWhe, the MA consumption of 3.3 Kg/TWhe.

The detail of the consumption of MA (Am+Cm+Np) shows that americium is incinerated (4.3 Kg/TWhe), and curium is still produced (0.2 Kg/TWhe). This is due to the non-optimization of the spectrum: the ratio (capture/fission) is higher in a thermal spectrum and that does not promote the fission. The plutonium, americium and curium consumption rate in mass at equilibrium are respectively 32 %, 20 %, -1 %. Although slightly degraded, the reactivity coefficients remain within acceptable limits. The double objective that consists in stabilizing the plutonium inventory and minimizing the (Am+Cm) inventories could be satisfied by assuming a scenario with 36 % of APA PWRs.

Due to the slow degradation of the plutonium isotopic vector, the enrichment in ^{235}U ranges from 2.02 % to 3.86 % between the 1st and the 4th recycling, and the fraction of fissile plutonium varies from 64 % to 39 %.

One important result of the scenario is that the (Pu+Am+Cm) inventory in the cycle can be stabilized. The total amount of Plutonium in the cycle ranges from 230 to 260 tons, with 140-160 tons outside the reactor, and 100 tons under irradiation. In case of (Am+Cm) management the total amount of (Am+Cm) is 75 tons (to be compared with the results of table XII for the CORAIL (Pu+MA) case).

Concerning the safety parameters, the increase of ^{235}U reduces by less than 5 % the absolute value of the boron efficiency and Doppler coefficient. The introduction of Pu or MA in the core results in a decrease of the void coefficient. It has been verified that the void coefficient remains negative at the beginning and at the end of each recycle.

5.3 CONCLUSION ON IMF

IMF fuels could present an interest for once-through Pu consumption. However the fuel forms and/or the assembly design are new challenges and need significant development efforts. Also, the licensing for use in existing PWRs can be a potential difficulty.

It is important to note that the inevitable and significant build-up of minor actinides, indicates that there is little or no benefit in terms of repository criteria (Ref. 20).

Moreover, in view of the results shown in section 5.2, there is a general agreement, that further MA burning can only be achieved in a fast spectrum, e.g. in an ADS in tier 2. However, the recovery of the MA, or of the mixture of residual Pu and built-up MA, has not been studied in detail, in order to insure the feasibility of the successive step. As far as the final fuel composition at the end of the IMF irradiation (Pu isotopic vector and MA isotopic content), it should be verified if it will be suitable for further burning in a dedicated burner (with fast spectrum).

6. Other thermal reactor concepts to transmute TRU: BWR, HTR, JAERI concept with undermoderated lattice

6.1 HTRs

It is interesting to consider Pu burning in HTGR with respect to the corresponding performance of PWRs, in particular loaded with IMF (see chapter 5).

Relatively few detailed studies have been performed. However some fundamental characteristics and issues have been analyzed (see e.g. ref. 16). In particular, if the objective is both the minimization of the residual Pu and of the Minor Actinide production in the discharged fuel, the achievement of high burn-ups allows one to reach a Pu destruction rate between 50 and 80 % depending of the Pu initial composition.

The MA production ranges from 5 to 18 % of the burnt Pu. A typical example is given in the following table:

Pu-type	Pu destruction rate	MA-production as fraction of burnt Pu	Average burn-up GWd/t
1st generation Pu in GTMHR	74.4 %	10 %	645
1st generation Pu in PBR	72.0	9.3	643
2nd generation Pu in GTMHR	71.5	12.5	607
3rd generation Pu in GTMHR	63.9	15.8	520
3rd generation Pu in PBR	50.0	9.0	428

Limiting factors for the performance of HTGRs with respect to the objective indicated above can be found :

a) in the graphite temperature reactivity coefficient which can be positive for degraded Pu vectors (e.g. at the end of cycle) ;

b) fast fluence and maximum burn-up acceptable by the coated particles.

Several factors have to be accounted for in order to optimize the HTGR performance. In ref. 16 , it is shown in particular that there exists for each isotopic Pu-composition, an optimum Pu-loading that maximizes the burn-up and then minimize the Pu discharge despite a continuous increase of the MA discharge mass (see figure 20).

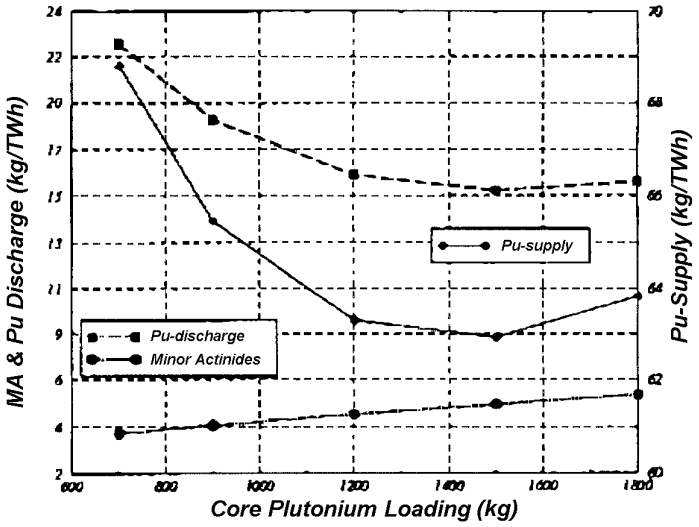


FIGURE 20 - Discharged masses & Pu-supply per TWh_{th} as a function of the initial Pu-loadings

6.2 LOW MODERATION LWRS

Low moderation LWRs have been proposed by JAERI (Japan). These would have high conversion ratios (in the region of 0.9 to 1.0). Plutonium is considered an important strategic resource that should be conserved for eventual use in fast reactors.

Studies have been performed on low moderation PWR and BWR fuel designs that achieve a very high conversion ratio through very low moderator/fuel volume ratios. The designs envisage a very tight lattice in which the spacing between fuel rods only marginally exceeds the fuel rod diameter, so that the water volume is greatly reduced.

Studies concentrate in particular on the thermal-hydraulic and mechanical design areas. The thermal-hydraulic area needs a concerted effort to establish new heat transfer correlations that would apply to these designs. Considerable mechanical design work would be needed to establish mechanical designs that are viable with the very narrow separation between fuel rods.

An advanced water-cooled reactor concept named Reduced-Moderation Water Reactor (RMWR) developed at JAERI in cooperation with JAPC and with the technical support from the Japanese LWR vendors, achieves a high conversion ratio over 1.0 with MOX fuel (Ref. 17). Both 1000 MWe and 300 MWe designs have been developed. A recent 300 MWe design is a double-flat-core design in a BWR-type core. In this design, the core water mass flow rate is significantly reduced to realize a high core average void fraction with a short fuel rod (<1 m). By these reasons, the pressure drop along the fuel assembly is evaluated to be around 30 kPa, and hence, the natural circulation cooling of the core is expected to be possible.

The core part is very short, i.e. 0.76 m high, and consists of two MOX regions and an internal blanket region between them. Adding the upper and lower blanket regions of 0.28 and 0.26 m high, the total axial length is 1.21 m. This very short double-flat-core design makes the void reactivity coefficient to be negative. However this point deserves a careful validation since a small value (negative or positive) of the void reactivity coefficient, results from the

compensation of large negative and positive contributions, normally affected by significant uncertainties.

The average fissile plutonium (Puf) content in the MOX regions is 18 % in this design, and the total Pu content is ~30%.

A full fuel cycle analysis is missing, and some work is in progress at ANL. This analysis should clarify the performance of the concept for a specific strategy. As it was previously indicated (section 3.1), the “natural” mission of a tight-pitch LWR core, is to preserve resources, in particular in view of a future deployment of fast reactors.

6.3 MOX IN BWRS

In many respects the technical issues associated with MOX fuels in BWRs are similar to those which arise in LWRs and VVERs, but there are some important distinctions. These arise from the very different fuel element used in BWRs and the fact the moderator/coolant water is a two phase mix of steam and water. BWRs use a shrouded fuel element in order that unvoided moderator can be maintained in the gaps between assemblies and they also incorporate internal water channels. The heterogeneity effects caused by the presence of inter-assembly water gaps and internal water channels necessitates the use of several enrichment regions even in a UO_2 assembly to reduce power peaking and BWR MOX assemblies need to use fuel rods with several different plutonium concentrations for the same reason. An important distinction between PWRs and BWRs is that BWRs do not use soluble boron in the moderator for reactivity control ; in BWRs the role of soluble boron for reactivity control is largely replaced by having UO_2/Gd_2O_3 (urania/gadolinia) burnable poison rods within each fuel element. The need for gadolinia burnable poison also applies to MOX fuel elements, and although it is possible in principle to incorporate gadolinia within MOX fuel rods, normal practice is to use urania/gadolinia rods. The requirement to minimise within-element power peaking leads to complicated fuel assembly designs with multiple plutonium concentrations and urania/gadolinia rods.

In principle, BWRs are in some ways more tolerant of MOX loading than PWRs, because the reactivity worth invested in BWR control rods tends to be higher (so that shutdown margins are less restrictive).

A study (ref. 18) has shown that there is potential for a 100 % MOX-BWR core to allow multiple Pu recycling. However, the possible range of Pu content depends on the Pu isotopic composition and on the Pu-fissile content. Solutions can be found for the reactivity control using enriched boron control rods, with an appropriate design. Once more, due to the global void effect limitation, the maximum number of recyclings is 3 to 4, when a ~ 35 % Pu-fissile content is reached.

7. A systematic intercomparison of concepts

The results given in the previous sections, enable to open multiple perspectives on the potential of TRU transmutation in LWRs, based on the homogeneous or heterogeneous recycling in relatively standard PWR assemblies, in particular in order to manage (stabilize, or increase or decrease) the TRU inventories. If the objective is to significantly reduce the TRU inventories, and in particular the Pu inventory, more innovative LWR concepts have to be introduced, as indicated in the previous discussion.

In order to gain an insight of the relative merits of the different approaches, it is more meaningful and effective to intercompare on physics ground the characteristics of different systems dedicated to the transmutation of Pu and Minor Actinides (MA), instead of comparing the performances of more or less sophisticated preliminary designs. The first attempt in this direction, was made in Ref. 19.

That analysis showed that fast neutron spectrum systems have an advantage in terms of neutron balance over thermal neutron systems. That analysis can be generalized and it is proposed to intercompare the following parameters:

- "D" (neutron consumption/fission) parameters for actinides.
- Neutron "surplus" in the core neutron balance.
- The ratio of the production of MA to the destruction of Pu+MA (at equilibrium), normalized to the generated power (expressed in TWhe).
- Compositions at equilibrium.
- βeff.
- The neutron source at fuel discharge and at fuel refabrication.
- The decay heat at fuel refabrication.
- The decay heat of wastes sent to the repository, at different times after storage (e.g. 100 years, 200 years etc).

These parameters account for characteristics of relevance both for the transmutation core, the fuel cycle and the impact on the repository. The source of potential radiotoxicity in the repository has not been explicitly introduced in this list, since for transmutation concepts which require multiple processing, it has been shown that the radiotoxicity discharged to waste is essentially dependent on the performance of the separation process, i.e. essentially on the assumed decontamination factors and consequent losses.

The proposed intercomparison will make use of generalized expressions for the neutron balance and the equilibrium state characteristics. It will also be attempted to characterize the reactivity response of the different systems to perturbation (e.g. coolant void coefficient) by some parameter, related, e.g. to the flatness of the adjoint function, or to some other characteristics, like the absorption rate in Pu-240 first resonance.

The systems which will be considered are:

- Standard PWRs, recycling Pu or TRU, with variable moderator-to-fuel ratio R ($1.4 \leq R \leq 4$).
- BWRs and SCWRs, recycling Pu or TRU.
- U-free (or "IMF") concepts, as indicated, e.g. in Ref. 14, and including the CONFU concept developed at MIT [15].

8. Conclusions

The present analysis has focused on some basic characteristics of Pu and TRU recycling in LWRs.

Pu separation and one recycle in LWRs has been demonstrated and is operational in different countries. As far as multirecycling, the feasibility of Pu multirecyclig has been demonstrated, and several realistic approaches can be envisaged. MA have never been recycled in LWRs, but the feasibility of their multirecycling does not present major obstacles from the reactor point of view, even if penalties (e.g. in terms of overenrichment in U-235) are expected. There are practical, and very important, difficulties related to the fuel cycle (in particular at fuel fabrication) , which could preclude a full MA multirecycling.

The different options here examined allow a flexible Pu management, from the stabilization of the Pu stocks, to their moderate decrease or even increase.

If a more drastic reduction is required, some new concepts have been proposed and new fuel forms have to be envisaged. Their development, validation and licencing will take time, and their deployment cannot be realistically foreseen in the short term.

As far as MA, the heterogeneous recycling of Am will not bring any significant gain in terms of radiotoxicity reduction, but potentially some gain in terms of heat load in the repository. This should be confirmed with a deeper analysis, and in any case the strategy for the storage, further processing or disposal of the residual TRU in the irradiated targets should be carefully evaluated. In fact, in this report we have underlined the difficulties associated to any Cm management strategy.

Finally, we have indicated a general and systematic approach, based on simple and well defined physics concepts, which will allow an effective intercomparison of concepts, in particular in the field of more innovative reactor/fuel cycle conceptual proposals. This approach will be the subject of a future investigation.

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